UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

November 8, 2005

Joseph E. Venable
Vice President Operations
Waterford 3
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000382/2005004

Dear Mr. Venable:

On September 26, 2005, the NRC completed an inspection at your Waterford Steam Electric Station, Unit 3. The enclosed report documents the inspection findings which were discussed on October 4, 2005, with you and other members of your staff.

This 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. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC identified three issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations were associated with these issues. These violations are being treated as noncited violations, consistent with Section VI.A of the Enforcement Policy. These findings are described in the subject inspection report. If you contest the subject or severity of a noncited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Waterford Steam Electric Station, Unit 3, facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be made 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).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

David N. Graves, Chief
Project Branch E
Division of Reactor Projects

Docket: 50-382
License: NPF-38

Enclosure:
NRC Inspection Report
050000382/2005004
w/attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-382
License No.:	NPF-38
Report No.:	50-382/05-004
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	June 27 through September 26, 2005
Inspectors:	M. Hay, Senior Resident Inspector, Waterford 3 G. Larkin, Resident Inspector G. George, Reactor Inspector J. Kirkland, Project Engineer D. Overland, Reactor Inspector P. Elkmann, Emergency Preparedness Inspector T. Hoeg, Senior Resident Inspector, St. Lucie Nuclear Plant V. Gaddy, Senior Project Engineer M. Haire, Operations Engineer R. Barkley, Senior Reactor Engineer M. Brown, Operations Engineer S. Garchow, Operations Engineer T. Farnholtz, Senior Project Engineer R. Kahler, Emergency Preparedness Inspector J. Drake, Operations Engineer
Approved By:	David N. Graves, Chief, Project Branch E
ATTACHMENTS:	Supplemental Information

Enclosure

SUMMARY OF FINDINGS

IR05000382/2005-004; 06/27/2005 - 09/26/2005; Waterford Steam Electric Station, Unit 3; Refueling Outage, Temporary Plant Modifications, and Problem Identification and Resolution

The report covered a 13-week period of inspection by resident inspectors and a regional emergency preparedness inspector. In response to hurricane Katrina, additional NRC inspectors from Regions I, II, and IV provided onsite event response coverage. The inspectors identified three Green findings. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the Significance Determination Process 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 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- SL-IV. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.59 for the failure to obtain NRC approval prior to implementing a change to the facility that resulted in a departure from a method of evaluation described in the final safety analysis report used in establishing the design bases. Specifically, the licensee implemented a change that assumed the unprotected dry cooling towers would not be impacted during a tornado event. This change was implemented based on the inappropriate use of a Tornado Missile Risk Evaluation method of evaluation not previously approved by the NRC. The licensee implemented this change to compensate for a licensee identified analysis error that adversely affected the ultimate heat sink capability following a tornado event. The licensee entered this deficiency into their corrective action program for resolution. The cause of this finding is related to the crosscutting element of human performance.

  The finding is greater than minor in that it affected the mitigating systems cornerstone attribute of equipment availability and function during a design bases tornado event. Regional and NRR staff determined that the change made by the licensee resulted in a departure from a method of evaluation described in the final safety analysis report used in establishing the design bases and that the change would require NRC approval under 10 CFR 50.59 guidance. In accordance with the NRC Enforcement Manual, violations of 10 CFR 50.59 are not processed directly through the significance determination process. Therefore, this issue was considered applicable as traditional enforcement. Although the significance determination process is not designed to assess significance of violations that potentially impact or impede the regulatory process, the technical result or condition of a 10 CFR 50.59 violation can be assessed through the significance determination process. The inspectors and the Region IV reactor analyst discussed the significance of this finding. A significance Determination Process Phase 1 screening was performed and the finding was determined to have very low safety significance because there was no actual loss of mitigating system safety function per Generic Letter 91-18 guidance. (Section 4OA2)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Green noncited violation of Technical Specification 6.8, “Procedures and Programs,” for the failure to establish adequate procedures regarding containment closure following loss of shutdown cooling while in reduced reactor coolant inventory conditions. This deficiency could result in loss of the containment barrier when called upon and the failure to maintain occupational radiation exposures as low as reasonably achievable. The licensee entered this deficiency into their corrective action program for resolution. The cause of this finding is related to the crosscutting element of human performance.

  The failure to establish adequate procedures for containment closure in reduced reactor coolant inventory conditions is greater than minor in that if left uncorrected the finding would become a more significant safety concern that could result in the loss of the containment barrier when called upon and the failure to maintain occupational radiation exposures as low as reasonably achievable. Using Manual Chapter 0609, Appendix H, “Containment Integrity Significance Determination Process,” the finding was assessed as a Type B finding. Through interviews and review of additional analysis the licensee provided reasonable assurance that following a loss of shutdown cooling containment closure would be performed prior to core uncovery with leakage less than 100 percent containment volume per day through the equipment hatch. Using MC 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process,” the licensee’s three year rolling average collective dose was less than 135 person-rem. Based on these assessments the finding was determined to be of very low safety significance. (Section 1R20)

- Green. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to preclude recurrence of through wall pipe leakage on the main steam line pipe 2MS2-123. This deficiency resulted in an unisolable steam leak requiring NRC approval to deviate from American Society of Mechanical Engineers Boiler and Pressure Vessel Code Case N523-2 to perform temporary repairs preventing a plant shutdown. The licensee entered this deficiency into their corrective action program for resolution. The inspectors determined the cause of this finding was related to the problem identification and resolution crosscutting area.

  The finding is greater than minor because it affected the reactor safety barrier integrity cornerstone for providing reasonable assurance that the physical design barriers protect the public from radionuclide releases caused by accidents or events. The finding was of very low safety significance because it did not result in an actual open pathway affecting the physical integrity of reactor containment. (Section 1R23)

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status: The plant began the period on June 27, 2005, at 100 percent power and remained at 100 percent power until August 28, 2005, when the plant was shutdown in anticipation of hurricane strength winds affecting the plant from Hurricane Katrina. On September 13, 2005, operators commenced a reactor startup and reached approximately 100 percent reactor power on September 14, 2005. Power remained at that level for the remainder of the inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed the licensee's preparations for Hurricane Dennis and Hurricane Katrina. The review included an evaluation of the licensee's operating experience, compensatory measures, and response procedures. The inspectors also completed walkdowns of plant grounds and risk significant systems necessary for plant shutdown.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors performed the following three partial system equipment alignment inspections during this inspection period. The inspectors performed a walkdown of accessible portions of these systems assessing material condition, housekeeping issues, and system configuration:

- On June 30, 2005, the inspectors performed a partial equipment alignment inspection of component cooling water system, Train B, while emergent maintenance activities were performed on the redundant train. System configuration was assessed using Operating Procedure OP-002-003, "Component Cooling Water," Revision 14.

- On July 27, 2005, the inspectors completed a partial equipment alignment inspection of component cooling water system, Train A. System configuration was assessed using Operating Procedure OP-002-003, "Component Cooling Water," Revision 14, following a train outage for maintenance.

- On August 17, 2005, the inspectors completed a partial equipment alignment inspection of containment spray system, Train A, while planned maintenance activities were performed on the redundant train. System configuration as assessed using Operating Procedure OP-903-034, "Containment Spray Valve Lineup Verification," Revision 5.

b. Findings

No Findings of significance were identified.

.2 Complete System Walkdowns

a. Inspection Scope

The inspectors performed a complete equipment alignment inspection of the high pressure safety injection system. A walk down of the mechanical and electrical components in the system was performed to verify that the system was configured and operated in accordance with operating procedures. The inspectors reviewed the system design requirements in the Updated Final Safety Analysis Report to verify the system's ability to perform its safety function for design basis events. The inspectors reviewed applicable design documentation and selected condition reports (CR) to verify that degraded conditions were identified at the appropriate threshold and that corrective actions were adequate and implemented in a timely manner.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors conducted six inspections to assess whether the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capabilities, and maintained passive fire protection features in good material condition.

The following areas were inspected:

- Fire Zones RAB 25, 32 and 39 on July 13, 2005
- Fire Zone RAB 8A, on July 30, 2005
- Fire Zones RAB 33, 35, 36, 37, 38 and 39 on August 2, 2005

- Fire Zones RAB 1A, 1C, 8B, 16, Wet Cooling Tower A, and RAB Roof East on September 14, 2005

- Fire Zones RAB 8C, 11, 12, 32 and RAB Roof West on September 16, 2005

- Fire Zones Turbine Generator Building +15-foot mean sea level and +40-foot mean sea level, RAB 1A, 1C, and Dry Cooling Tower A on September 9, 2005

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors evaluated external flood protection measures to ensure that flood risks were adequately mitigated. The inspectors reviewed the Updated Final Safety Analysis Report, Procedure OP-901-521, "Severe Weather and Flooding," and Calculation EC-C91-015, "Evaluation of NRC Generic Letter 89-22, Revised PMP Criteria." The inspection included a review of water tight doors and piping penetrations below the +30-foot mean sea level and the fuel handling building roof drainage system.

b. Findings

No findings of safety significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On July 12, 2005, the inspectors observed a licensed operator simulator training scenario. During the scenario, operators responded to problems associated with multiple instrument failures, loss of a safety bus, and a loss of coolant accident with failure of a containment spray pump to automatically start. The simulator training evaluated the operators' ability to recognize, diagnose, and respond to abnormal and emergency reactor plant conditions. The inspectors observed and evaluated the following areas:

- Understanding and interpreting annunciator and alarm signals

- Verifying automatic actions and analyzing plant parameters in abnormal and emergency conditions

- Use and adherence of Technical Specifications (TS)

- Communicating as a team and prioritizing actions with attention to detail

- The crew's and evaluator's critiques

- Classifying emergencies and making notifications

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

During the inspection period, the inspectors reviewed Entergy's implementation of the Maintenance Rule. The inspectors considered the characterization, safety significance, performance criteria, and the appropriateness of goals and corrective actions. The inspectors assessed Entergy's implementation of the Maintenance Rule to the requirements outlined in 10 CFR 50.65 and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2. The inspectors reviewed the following two Maintenance Rule scoped systems, structures, or components that displayed performance problems:

- Chemical and volume control system
- Auxiliary component cooling water system

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors reviewed risk assessments for planned or emergent maintenance activities to determine if the licensee met the requirements of 10 CFR 50.65(a)(4) for assessing and managing any increase in risk from these activities. The following four risk evaluations were reviewed:

- On July 12, 2005, during planned maintenance and testing on safety injection and containment spray system recirculation header, Train B, isolation Valves SI-120B and -121B

- On July 14, 2005, while placing the steam bypass control system in manual operation for emergent troubleshooting and repair activities

- On August 18, 2005, while preforming emergent troubleshooting and repair activities on Train B essential chiller

- On September 29, 2005, during planned maintenance activities on Train A emergency diesel generator

b. Findings

No findings of significance were identified.

1R14 Non-Routine Evolutions and Events (71111.14)

a. Inspection Scope

On August 27, 2005, Entergy declared a Notice of Unusual Event due to a hurricane warning. In preparation for the hurricane Entergy performed a plant shutdown on August 28, 2005. On August 29, 2005, Entergy declared the offsite distribution system inoperable at 6:24 a.m. due to system voltage being greater than allowed by TS. At 7:59 a.m., that same morning, a complete loss of offsite power occurred resulting in an automatic start of both emergency diesel generators.

Prior to the hurricane the NRC took steps to be prepared for actions in the event the hurricane impacted the facility. The NRC entered the Monitoring Mode and activated the Region IV Incident Response and Headquarters Operations Centers on August 28, 2005, at 4:00 p.m. Two regional inspectors were sent to the facility to provide continuous monitoring of activities. Prior to, during, and following the event the inspectors verified that plant evolutions were performed in accordance with plant TSs and implementing procedures. These activities included a review of licensee actions in preparation for adverse weather and potential flooding and direct observations of operator performance during the loss of offsite power event.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the technical adequacy of four operability evaluations to verify that they were sufficient to justify continued operation of a system or component. The inspectors considered that, although equipment was potentially degraded, the operability evaluation provided adequate justification that the equipment could still meet its TS, Updated Final Safety Analysis Report, and design bases requirements and that the potential risk increase contributed by the degraded equipment was thoroughly evaluated. The following four evaluations were reviewed:

- Operability evaluation addressing air pressure regulator Valve EGA-417B having an air leak causing downstream pressure to be maintained at pressure greater than normal (CR-WF3-2005-03127)
- Operability evaluation addressing a non-conservative assumption affecting the design basis tornado event analysis (CR-WF3-2005-3431)
- Operability evaluation addressing a large flow imbalance between component cooling water system Trains A and B due to an abnormal valve alignment to support Valve CC-200B maintenance (CR-WF3-2005-03577)
- Operability evaluation addressing a decrease in containment spray riser level during refill operations (CR-WF3-2005-03533)

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds (71111.16)

a. Inspection Scope

The inspectors completed one evaluation of the effect of one operator workaround. The inspectors also reviewed the third quarter 2005 Watchstation Deficiency List and assessed the effect of the workarounds on the ability of operators to implement plant emergency operating procedures. The inspectors completed the review to verify that the cumulative effect of workarounds did not challenge the operators’ capability to respond to plant transients and events. The inspectors completed an inspection of an operator workaround involving a degraded pressurizer level indication required for safe shutdown from the alternate shutdown panel. The degraded condition resulted in deviation between the two pressurizer level channels requiring operators to frequently monitor the divergence and take actions to maintain the channel operable. These actions consisted of degassing the pressurizer and reference leg fills of the pressurizer level instrument to maintain the instrument operable.

a. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed postmaintenance tests to verify system operability and functional capabilities. The inspectors considered whether testing met design and licensing bases requirements, TSs, and licensee procedural requirements. The inspectors reviewed the testing results for the following five components:

- Component cooling water Valve CC-135B, following repairs on July 23, 2005
- Control room emergency filtration unit recirculation Damper HVC-213A following maintenance activities performed on July 27, 2005
- Train A emergency diesel generator following emergent repairs on September 1, 2005
- Dry cooling tower Fan 9B following emergent repairs on September 5, 2005
- Safety injection system Valve SI-335B following emergent repairs on September 12, 2005

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

On August 28, 2005, Entergy performed a normal reactor shutdown in preparation for Hurricane Katrina. The facility remained in a shutdown condition through September 13, 2005, when they performed a reactor startup. During the outage, the inspectors observed shutdown, cooldown, startup, and maintenance activities to verify that the licensee maintained plant capabilities within the applicable TSs requirements. Specific performance activities evaluated include:

C Reactor Water Inventory Controls - to verify flow paths, equipment configurations, and alternative means for inventory addition were appropriate to prevent inventory loss

- Reactivity Controls - to ensure compliance with TSs and to verify activities which could affect reactivity were properly controlled
- Containment Closure - to verify that the licensee controlled containment penetrations in accordance with TSs and procedural requirements
- Shutdown Cooling System - to verify that operating parameters were established and maintained within the required range

- Reactor Coolant System Instrumentation - to verity that reactor coolant system pressure, level, and temperature instrumentation were installed, configured, and maintained to provide accurate indication

- Heatup and Startup Activities - to ensure that TSs and administrative procedure prerequisites for mode changes were met prior to changing modes or plant configurations

b. Findings

Introduction. The inspectors identified a Green noncited violation (NCV) of TS 6.8, “Procedures and Programs,” for the failure to establish adequate procedures regarding containment closure following loss of shutdown cooling while in reduced reactor coolant inventory conditions.

Discussion. The inspectors reviewed the licensee’s containment closure criteria for reduced reactor coolant inventory conditions. Operations Procedure OP-901-131, “Shutdown Cooling Malfunction,” directs operators to complete containment closure actions within one hour following a loss of shutdown cooling with the reactor coolant system less than 18 feet mean sea level, approximately 2 feet below the reactor head flange. The inspectors noted that on April 22, 2005, during mid-loop operations five days following shutdown, time to boil following a loss of shutdown cooling would be approximately 18 minutes and the containment closure time criteria was set at one hour.

The inspectors reviewed engineering analysis CE NPSD-616, “Alternate Containment Closure Criterion During Loss of Shutdown Cooling at Mid-Loop Conditions.” This analysis evaluated the containment environmental conditions following a loss of shutdown cooling event. The inspectors noted that the analysis was based on containment closure 20 minutes following loss of shutdown cooling, not 60 minutes as procedurally allowed, therefore the analysis results were not conservative with respect to the containment closure procedure. The analysis concluded that following boiling airborne radioactivity in containment can build up quite rapidly. The Iodine-131 concentration calculated at the time of containment closure was approximately 500 DAC assuming a reactor coolant activity of 0.1 micro-curies/gram. The inspectors noted that this value was not assessed 60 minutes following a loss of shutdown cooling which by procedure is the maximum allowed time to close containment, and also noted that operations procedure OP-010-006, “Outage Operations,” allowed a reactor coolant activity level of 0.240 micro-curies/gram. The inspectors were informed that this activity level was calculated based on evacuation of personnel from containment during the first half hour of the event producing a dose equivalent of 500 DAC hours (1.25 rem). The inspectors noted that this analysis assumed all personnel had exited containment in the first half hour following loss of shutdown cooling although by procedure, licensee staff could be in containment for as long as one hour.

In discussions with radiation protection management the inspectors discussed what provisions would be made to ensure the personnel directed to close the equipment hatch would be adequately informed and protected from the expected radiological conditions. Radiation protection management stated they were unaware that radiological hazards would be present and had no knowledge that an analysis had been performed to evaluate the containment closure activity. The licensee stated they would be evaluating these deficiencies.

The inspectors noted that analysis CE NPSD-616, "Alternate Containment Closure Criterion During Loss of Shutdown Cooling at Mid-Loop Conditions," evaluated the temperature and pressure effects inside containment following a loss of shutdown cooling that resulted in reactor coolant system (RCS) boiling. The analysis concluded that a sudden change in the rate of temperature and pressure increase would occur at the time of containment closure. The inspectors determined that this clearly implied that much of the energy prior to closure would be passing through the equipment hatch. The inspectors questioned the licensee if they had evaluated the ability to close the equipment hatch taking into account that as the hatch cover is moved to a closed position the flow area will decrease resulting in a velocity increase that could introduce forces against the hatch making it difficult to control closure. The licensee stated they had not evaluated this concern.

After discussing these observations the licensee agreed that enhanced procedural guidance was necessary and entered these deficiencies into the corrective action program as CR-WF3-2005-03763. The licensee planned to resolve these deficiencies prior to future opening of the equipment hatch during Mid-Loop operations.

Analysis. The failure to establish adequate procedures for containment closure in reduced reactor coolant inventory conditions is greater than minor in that if left uncorrected the finding would become a more significant safety concern that could result in the loss of the containment barrier when called upon and the failure to maintain occupational radiation exposures ALARA. The inspectors and a Region IV reactor analyst discussed the significance of this finding. Using Manual Chapter (MC) 0609, Appendix H, "Containment Integrity Significance Determination Process," the finding was assessed as a Type B finding. Through interviews and the review of additional analyses, the inspectors concluded that the licensee provided reasonable assurance that following a loss of shutdown cooling containment closure would be performed prior to core uncovery with leakage less than 100 percent containment volume per day through the equipment hatch. Also, using MC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," the licensee's three year rolling average collective dose was less than 135 person-rem. Based on these assessments the finding was determined to be of very low safety significance. The inspectors determined the cause of this finding is related to the crosscutting element of human performance.

Enforcement. Technical Specification 6.8.1.a requires, in part, that written procedures be established, implemented, and maintained covering the activities recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33 recommends procedures for combating emergencies such as a loss of shutdown cooling event. The failure to establish adequate procedures to ensure containment closure following a loss of shutdown cooling event is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy, and is identified as NCV 05000382/2005004-01: Failure to Establish Adequate Procedure for Containment Closure Following a Loss of Shutdown Cooling Event. This issue is in the licensee's corrective action program as CR-WF3-2005-03763.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed or reviewed the following four surveillance tests to ensure the systems were capable of performing their safety function and to assess their operational readiness. Specifically, the inspectors considered whether the following four surveillance tests met TSs, the Updated Final Safety Analysis Report, and licensee's procedural requirements:

- Surveillance Procedure OP-903-100, "MOV Overload Bypass Test," Revision 7, performed on July 12, 2005. This surveillance verified functional capabilities of the thermal overload bypass circuitry for safety injection and containment spray system recirculation header Train B isolation Valves SI-120B and -121B.

- Startup Test Procedure NE-002-002, "Variable Tavg Test," Revision 12, performed on July 15, 2005. This test verifies that the moderator temperature coefficient is within its limits specified by the core operating limits report.

- Surveillance Procedure OP-903-050, "Component Cooling Water and Auxiliary Component Cooling Water Pump and Valve Operability Test," Revision 17, performed on July 30, 2005. This surveillance verified operability of Auxiliary Component Cooling Water Pump A.

- Surveillance Procedure OP-903-130, "Verification of Locked Valves and Breakers," Revision 2, performed on September 10, 2005. This surveillance verifies that each component required to be locked was locked and properly positioned.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed temporary plant modification, ER-W3-2005-0433-000, “Leak on Line 2MS2-123, Downstream of 2MS12-47B, Drip Pot Upstream of Main Steam Isolation Valve Number 2.” This modification installed a mechanical clamping device to temporarily repair a through wall pipe leak that was unisolable from steam generator Number 2. The temporary modification was a deviation from ASME Code Case N-523-2, “Mechanical Clamping Devices for Class 2 and 3 Piping,” requiring NRC approval. The deviation allowed using this repair method for piping that formed a part of the containment boundary. This deviation also prevented not requiring plant shutdown in order to perform the repair. The inspectors reviewed the safety screening, design documents, UFSAR, and applicable TSs to determine that the temporary modification was consistent with the modification documents, drawings, and procedures. The inspectors walked down the accessible portions of the affected equipment.

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to preclude recurrence of through wall pipe leakage on the main steam line Pipe 2MS2-123.

Description. On September 14, 2005, a through wall leak was identified on safety related main steam drain Pipe 2MS2-123. The licensee determined that the steam leak was a result of external corrosion. The piping was exposed to outside environmental conditions resulting in water soaking the pipe insulation causing the carbon steel piping to corrode. Through review of the licensee’s corrective action process the inspectors discovered additional examples of external corrosion resulting in leaks for similar piping exposed to outside environmental conditions. On March 7, 1997, Entergy identified that a drain line through wall leak developed from rain water that soaked the pipe insulation causing the carbon steel pipe to corrode. On November 26, 1999, external corrosion resulted in a through wall leak on drain Line 2MS2-123, the same segment of piping as the leak identified on September 14, 2005. Repairs for the 1999 leak resulted in a plant shutdown because the leak was unisolable. On August 20, 2005, the licensee identified a drain line through wall leak due to external corrosion downstream of main steam isolation valve Number 2.

Following the leak identified in November 1999, Entergy determined that corrective actions taken following the leak identified in March 1997 were not adequate to prevent recurrence. Specific root causes identified by the licensee in 1999 included; 1) piping insulation was found saturated with water; 2) piping was not properly coated in accordance with design specifications; and 3) no program was procedurally established to inspect for external corrosion. Additionally, in 1997 the licensee evaluated the feasability of replacing carbon steel pipe susceptible to external corrosion with stainless steel piping. The licensee determined that the additional costs associated with analysis and the design change would outweigh the benefit of replacing the material.

The inspectors determined that the failure to implement effective corrective actions in 1997 and 1999 resulted in the unisolable steam leak of safety related main steam Pipe 2MS2-123 identified in September 2005. The failure in 2005 was a result of wetted insulation causing external corrosion of carbon steel piping. The coating applied in 1999 was found to not provide protection because it did not adhere to the piping and the licensee had not inspected the piping since the previous failure of the same pipe in 1999.

Analysis. The deficiency associated with this finding was the failure to preclude recurrence of through wall pipe leakage on the main steam Line 2MS2-123. The finding is greater than minor because it affected the reactor safety barrier integrity cornerstone for providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The finding was evaluated using Inspection Manual Chapter 0609, Significance Determination Process, Appendix A for Initiating Events, Mitigating Systems, and Barrier Cornerstones. The finding was of very low safety significance because it did not result in an actual open pathway affecting the physical integrity of reactor containment. The inspectors determined the cause of this finding impacted the problem identification and resolution crosscutting area.

Enforcement. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion XVI, states in part, that, measures shall be established to assure that conditions adverse to quality, such as failures, are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The failure to preclude recurrence of through wall pipe leakage on main steam Pipe 2MS2-123 is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program as CR-WF3-2005-3983, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000382/200504-02, Failure to Prevent Recurrence of Main Steam Line Through Wall Pipe Leakage.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector performed an in-office review of Revision 31, Change 1, to the Waterford 3 Emergency Plan, and Revision 20, Change 2, to Procedure EP-001-001, "Recognition and Classification of Emergency Conditions," both submitted June 13, 2005. These revisions:

- Updated references to the reactor core output to its current value
- Deleted a table of exclusion area boundary calculated exposures
- Added references to the table of exclusion area boundary calculated exposures in the Updated Final Safety Analysis Report
- Changed 22 effluent release monitor values used in Emergency Action Levels A/UE/I, A/A/I, A/SAE/I, A/A/II, A/SAE/II, and A/SAE/III

These revisions were compared to their previous revisions, to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, and to the requirements of 10 CFR 50.47(b) and 50.54(q) to determine if the licensee adequately implemented 10 CFR 50.54(q).

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Daily Review

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for followup, the inspectors performed daily screening of items entered into the licensee's corrective action program. This was accomplished by reviewing the description of each new CR.

.2 Annual Sample Review

a. Inspection Scope

The inspectors reviewed licensee actions to resolve problems associated with identifying a non-conservative assumption affecting the design basis tornado event analysis. This review began as a look at how the licensee addressed the problem associated with performing analysis revisions. However, due to the significant impact of the identified discrepancy the licensee implemented a change to the facility as described in the Final Safety Analysis Report. The inspectors reviewed the adequacy of this corrective action.

b. Findings and Observations

Introduction. The inspectors identified a Severity Level IV NCV for the failure to obtain NRC approval prior to implementing a change to the facility that resulted in a departure from a method of evaluation described in the FSAR used in establishing the design bases.

Description. The inspectors reviewed CR-WF3-2005-01247. The report described a condition involving the licensee identifying that calculation MN(Q)-9-17, "Tornado Multiple Missile Protection of Cooling Towers," contained a non-conservative assumption. The non-conservative assumption involved lower component cooling water flow rates than would actually occur during the design basis tornado event following initiation of shutdown cooling. Higher flow rates would result in less efficient heat transfer through the dry cooling towers (DCT) that are part of the ultimate heat sink. Based on this discovery, the time after shutdown that the DCT could support shutdown cooling operations was premature, and emergency feedwater operations would be required for longer than originally assumed. The use of emergency feedwater for this additional time after shutdown would deplete the available inventory sources of water to mitigate this event.

The inspectors reviewed the design basis tornado event scenario for the facility. Final Safety Analysis Report Section 9.2.5, "Ultimate Heat Sink," states, in part, that the dry cooling towers have been designed with multiple cells (five cells for each dry cooling tower), and multiple fans (15 fans for each dry cooling tower), to ensure that damage by tornado missiles would not significantly affect heat removal capability. In addition to the multiple cells and fans which reduce the probability of damage to the cooling towers by tornado missiles, 60 percent of the dry cooling tower coils have been protected by grating located above them and designed to withstand tornado missiles. Sixty percent of the dry cooling towers will provide sufficient heat dissipation to atmosphere and will assure safe shutdown of the unit after a design basis tornado. The inspectors reviewed Waterford 3 Safety Evaluation Report, Section 9.2.5, "Ultimate Heat Sink," contained in NUREG-0787, dated July 1981. This report states, "The applicant has shown by analysis that sufficient heat removal capability is provided for 24 hours to maintain plant safety and assure safe shutdown assuming only 60 percent of the dry towers is available plus the water volume in the wet tower basins and assuming the most limiting single failure coincident with loss of offsite power."

The inspectors noted that the licensee resolved the identified analysis deficiency by changing the assumption that the unprotected dry cooling towers would fail during a tornado event. The licensee determined that since the probability of a missile strike was acceptably low, failure of the unprotected towers would not be assumed during a tornado event. This change provided additional dry cooling tower heat removal capacity needed to address the original analysis error.

The inspectors reviewed the licensee's 10 CFR 50.59 screening for this change. The licensee determined that a 50.59 evaluation was not required based on inappropriately assuming that the change did not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses of the facility. The inspectors determined this error was a result of an inappropriate interpretation of Waterford Steam Electric Station, Unit 3, Licensing Amendment 168 dated September 7, 2000. This amendment to the facility operating license revised FSAR Section 2.3.1.2.4, “Tornadoes,” with regard to design requirements for physical protection from tornado missiles. Specifically, this amendment allowed the use of a TORMIS methodology to demonstrate that specific plant features that are currently unprotected at Waterford 3 would not require additional missile protection barriers due to the low probability of a tornado missile strike. The inspectors noted that this amendment did not allow use of the TORMIS methodology to change the original design basis that assumed the unprotected dry cooling towers would not be available following a tornado event.

Analysis. The inspectors referred to MC 0612, Appendix B, and determined that the finding is greater than minor in that it affected the mitigating systems cornerstone attribute of equipment availability and function during a design bases tornado event. Regional and NRR staff determined that the change made by the licensee resulted in a departure from a method of evaluation described in the FSAR used in establishing the design bases and that the change would require NRC approval under 10 CFR 50.59 requirements. The failure to request approval prior to the change is considered a violation of NRC requirements. In accordance with the NRC Enforcement Manual, violations of 10 CFR 50.59 are not processed directly through the significance determination process. Therefore, this issue was considered applicable as traditional enforcement.

Although the significance determination process is not designed to assess significance of violations that potentially impact or impede the regulatory process, the technical result or condition of a 10 CFR 50.59 violation can be assessed through the significance determination process. The inspectors and the Region IV reactor analyst discussed the significance of this finding. A significance Determination Process Phase 1 screening was performed and the finding was determined to have very low safety significance because there was no actual loss of mitigating system safety function per Generic Letter 91-18 guidance. The inspectors determined the cause of this finding is related to the crosscutting element of human performance.

Enforcement. 10 CFR 50.59, states, in part, that a licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses. Contrary to this, the licensee implemented a change that assumed the unprotected dry cooling towers would not be impacted during a tornado event. This change was implemented based on the inappropriate use of a TORMIS method of evaluation not previously approved by the NRC. Because this failure to comply with

10 CFR 50.59 requirements is of very low safety significance and has been entered into the corrective action program as CR-WF3-2005-3431, this violation is being treated as a noncited Severity Level IV violation, consistent with Section VI.A of the Enforcement Policy. This violation is identified as NCV 05000382/2005004-03, Change to a Method of Evaluation Without Prior NRC Approval.

.3 References to Issues with Crosscutting Aspects

Section 1R23 describes that the licensee failed to implement effective corrective actions to preclude recurrence of through wall pipe leakage on the main steam line pipe 2MS2-123. The cause of this finding is related to the crosscutting element of problem identification and resolution.

4OA3 Event Followup (71153)

.1 Hurricane Katrina Event

a. Inspection Scope

Following Hurricane Katrina, NRC inspectors and representatives of the Federal Emergency Management Agency assessed the readiness of Waterford 3 Steam Electric Station to restart following the plant shutdown that occurred on August 28, 2005, in preparation for the hurricane. This assessment was implemented in accordance with NRC MC 1601, “Communication and Coordination Protocol for Determining the Status of Offsite Emergency Preparedness Following a Natural Disaster, Malevolent Act, or Extended Plant Shutdown.” Specific activities included an assessment of both the onsite and offsite emergency preparedness capabilities and evaluating the readiness of the licensee’s systems, structures, components, and staffing levels to support safe plant operations.

To evaluate the condition of the onsite emergency preparedness infrastructure the inspectors determined the status of the following:

- Communication circuits between the licensee and the offsite authorities
- Licensee’s emergency response facilities
- Licensee’s ability to staff critical emergency response positions
- Environmental monitoring
- Meteorological monitoring
- Evacuation routes to and from the plant site
- Emergency sirens
- Structures, systems, and components needed for emergency action level classifications

Additionally, the inspectors reviewed the licensee’s assessment of emergency preparedness program deficiencies. For those deficiencies identified the inspectors reviewed documentation, interviewed personnel, and verified by observations that the compensatory measures in place were adequate.

b. Findings and Observations

After review the inspectors concluded that both the onsite and offsite emergency preparedness infrastructure were adequate to support plant restart and provide reasonable assurance that protective measures could be taken to protect public health and safety in the event of a radiological emergency. Additionally, the inspectors concluded that all systems, structures, components, and staffing levels were adequate to support safe operation of the plant following restart.

.2 (Closed) LER 05000382/2004-004-00: Failure to Adequately Size Containment Electrical Penetration Devices Due to Latent Personnel Errors

On November 17, 2004, the licensee identified that a 125 Vdc control power circuit did not have adequately sized over-current protection devices in accordance with TS 3.8.4.1a. This deficiency was identified during extent of condition review associated with a previous licensee identified condition reported in LER 05000382/2004-001-00 discussed in NRC Inspection Report 05000382/2004-04. This item is being considered an additional example of the licensee identified violation of TS 3.8.4.1a previously discussed. This LER is closed.

.3 (Closed) LER 05000382/2005-001-00: RCS Pressure Boundary Leakage Due to Primary Water Stress Corrosion Cracking of a Pressurizer Heater Sleeve

On April 19, 2005, during Refueling Outage 13, the licensee identified indications of reactor coolant system pressure boundary leakage from pressurizer heater Sleeves C-4 and D-2 due to primary water stress corrosion cracking. Occurrences of pressure boundary leakage due to primary water stress corrosion cracking were also identified in Refueling Outage 12 and reported in LER 05000382/2003-003-00. NRC Inspection Report 05000382/2003-03 determined that pressure boundary leakage identified in Refueling Outage 12 was the result of ineffective corrective actions resulting in a NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," NCV 05000382/2003007-04. Corrective actions to address this deficiency included a repair plan for all Alloy 600 small bore nozzles on the pressurizer and hot legs during Refueling Outage 13. Based on these corrective actions the pressure boundary leakage identified during Refueling Outage 13 is being considered an additional example of NCV 05000382/2003007-04. This LER is closed.

.4 (Closed) LER 05000382/2005-002-00: Failure of One System of Leakage Detection Instrumentation due to Latent Human Error

On June 9, 2005, during Refueling Outage 13, the licensee identified that the containment fan cooler condensate flow switches were inoperable for approximately the previous 3 years. The flow switches were determined to be inoperable due to their inability to detect a reactor coolant system leakage rate of 1 gpm in accordance with design requirements. The cause of the deficiency was the failure to incorporate the requirements and acceptance limits contained in applicable design documents into surveillance testing procedures. Corrective actions included revising and implementing the applicable surveillance procedure prior to exiting Refueling Outage 13 demonstrating the flow switches could detect a leakage rate of 1 gpm. This finding is more than minor because it affected the reactor coolant system leakage attribute of the Barrier Integrity cornerstone. The finding was of very low safety significance because other methods of reactor coolant system leak detection were available to the licensee and the unavailability of the flow switches did not contribute to an increase in core damage sequences when evaluated using the SDP Phase 2 worksheets. This LER is closed.

.5 (Closed) LER 05000382/2005-003-00: TS Minimum Volume Requirements in Diesel Generator (DG) Fuel Oil Storage Tank B Not Met Due to Bent Transmitter Tubing

On July 5, 2005, it was determined that Waterford 3 operated in a condition prohibited by TS. Technical Sspecification 3.8.1.1.b.2 requires a minimum volume of 39,300 gallons in the DG fuel oil storage tanks (FOST), or a fuel oil volume less than 39,300 gallons and greater than 37,000 gallons of fuel for a period not to exceed 5 days provided replacement fuel is onsite within 48 hours. Contrary to TS 3.8.1.1.b.2, Waterford 3 operated with DG FOST B at approximately 38,800 gallons from May 28, 2005 to June 16, 2005 resulting in not complying with TS Action b of TS 3.8.1.1 which requires restoration within 72 hours. The cause of this event was inadequate venting of trapped air due to the sensing line not maintained in a configuration continuously sloped from the instrument to the process pipe as required by design documentation. This design deficiency was previously dispositioned in NRC Inspection Report 05000382/2005003, Section 1R22, “Surveillance Testing,” as a Green NCV (NCV 05000382/2005003-01). This LER is closed.

4OA4 Crosscutting Aspects of Findings

Section 1R20 describes that the licensee failed to establish adequate procedural guidance for containment following loss of shutdown cooling during reduced reactor coolant inventory conditions. This deficiency could result in loss of the containment barrier when called upon and the failure to maintain occupational radiation exposures ALARA. The cause of this finding is related to the crosscutting area of human performance.

Section 4OA2 described that the licensee inappropriately interpreted a license amendment condition resulting in making a change to the facility without receiving prior NRC approval. The cause of this finding is related to the crosscutting area of human performance.

4OA6 Meetings

Exit Meeting Summary

.1 On August 3, 2005, the inspector conducted a telephonic exit meeting to present the inspection results to Mr. J. Lewis, Manager, Emergency Preparedness, who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

.2 The resident inspectors presented the inspection results to Mr. J. Venable, Vice President, Operations and other members of licensee management at the conclusion of the inspection on October 4, 2005. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a NCV.

- 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” requires that test procedures incorporate the requirements and acceptance limits contained in applicable design documents to ensure structures, systems, and components will perform satisfactorily in service. Contrary to this, Surveillance Procedure MI-003-409, “Containment Air Cooler Condensate Flow Switches Channel Functional Test CCS IFS 5160 A1, A2, B, C1, C2, and D,” failed to incorporate the applicable requirements and acceptance limits resulting in failure of the flow instruments to detect a reactor coolant system leakage rate of 1 gpm. This was identified in the licensee’s corrective action program as CR-WF3-2005-02412. This finding is of very low safety significance because other methods of reactor coolant system leak detection were available to the licensee and the unavailability of the flow switches did not contribute to an increase in core damage sequences when evaluated using the SDP Phase 2 worksheets.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Anders, Superintendent, Plant Security
C. Fugate, Assistant Manager, Operations (Shift)
T. Gaudet, Director, Planning and Scheduling
J. Lewis, Manager, Emergency Preparedness
T. Mitchell, Director, Engineering
R. Murillo, Senior Staff Engineer, Licensing
R. Osborne, Manager, Programs and Components
K. Peters, Director, Nuclear Safety Assurance
O. Pipkins, Senior Licensing Engineer
G. Scott, Licensing Engineer
J. Venable, Vice President, Operations
K. T. Walsh, General Manager, Plant Operations

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000382/2005004-01	NCV	Failure to Establish Adequate Procedure for Containment Closure Following a Loss of Shutdown Cooling Event
05000382/2005004-02	NCV	Failure to Prevent Recurrence of Main Steam Line Through Wall Pipe Leakage
05000382/2005004-03	NCV	Change to a Method of Evaluation Without Prior NRC Approval

Closed

05000382/2004-004-00	LER	Failure to Adequately Size Containment Electrical Penetration Devices Due to Latent Personnel Errors
05000382/2005-001-00	LER	RCS Pressure Boundary Leakage Due to Primary Water Stress Corrosion Cracking of a Pressurizer Heater Sleeve
05000382/2005-002-00	LER	Failure of One System of Leakage Detection Instrumentation due to Latent Human Error
05000382/2005-003-00	LER	TS Minimum Volume Requirements in DG Fuel Oil Storage Tank B Not Met Due to Bent Transmitter Tubing
05000382/2005004-01	NCV	Failure to Establish Adequate Procedure for Containment Closure Following a Loss of Shutdown Cooling Event
05000382/2005004-02	NCV	Failure to Prevent Recurrence of Main Steam Line Through Wall Pipe Leakage
05000382/2005004-03	NCV	Change to a Method of Evaluation Without Prior NRC Approval

LIST OF DOCUMENTS REVIEWED

Section 1R01: <u>Adverse Weather Protection</u>

Procedures

NUMBER	TITLE	REVISION
OP-901-521	Severe Weather and Flooding	4

Section 1R04: <u>Equipment Alignment</u>

Drawings

NUMBER	TITLE	REVISION
G160, Sh. 2	Component Closed Cooling Water System	48
G160, Sh. 3	Component Closed Cooling Water System	31
G160, Sh. 4	Component Closed Cooling Water System	13
G160, Sh. 5	Component Closed Cooling Water System	17
G160, Sh. 6	Component Closed Cooling Water System	12

Work Orders

WO 64836

Procedures:

NUMBER	TITLE	REVISION
OP-009-008	Safety Injection System	18
OP-903-030	Safety Injection Pump Verification	14
OP-903-011	High Pressure Safety Injection Pump Operability Check	9
OP-903-108	Safety Injection Balance Test	4

Condition Reports

50979745, 50232677

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
EC-M98-069	HPSI System Performance Surveillance Requirement Basis	1
W3-DBD-001	Safety Injection System	3

Section 1R05: Fire Protection

Procedure

NUMBER	TITLE	REVISION
UNT-005-013	Fire Protection Program	9
OP 009-004	Fire Protection	11-8
MM-007-010	Fire Extinguisher Inspection and Extinguisher Replacement	13
Administrative Procedure UNT-005-013	Fire Protection Program	9
Fire Protection Procedure FP-001-015	Fire Protection System Impairments	17
Training Manual Procedure NTP-202	Fire Protection Training	11-4

Section 1R06: Flood Protection Measures

Miscellaneous

NUMBER	TITLE	REVISION
EC-C91-015	Evaluation of Generic Letter 89-22, Revised PMP Criteria	0

OP-901-521	Severe Weather and Flooding	4

Condition Reports

CR-WF3-2004-00575
CR-WF3-2004-02438
CR-WF3-2005-00591
CR-WF3-2005-00954

Section 1R12: Maintenance Rule

Procedures:

NUMBER	TITLE	REVISION
MM-006-021	Charging Pump Maintenance	4
DC-121	Maintenance Rule	1

Condition Reports

CR-WF3-2004-1501	CR-WF3-2004-1280	CR-WF3-2004-1236
CR-WF3-2005-0332	CR-WF3-2004-1274	CR-WF3-2004-1224

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
ER-W3-2004-0250	Charging Pump AB Internal Check Valves	0
Regulatory Guide 1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	

Section 1R13: Risk Assessments

Procedure

NUMBER	TITLE	REVISION
CEP-IST-1	IST Bases Document	3

1R14: Non-Routine Evolutions and Events

Procedure

NUMBER	TITLE	REVISION
EP-001-001	Recognition and Classification of Emergency Conditions	20
EP-001-010	Unusual Event	24
OP-901-521	Severe Weather and Flooding	4
OP-902-003	Loss of Offsite Power/Loss of Forced Circulation Recovery	5
OP-010-005	Plant Shutdown	4

Section 1R15: Operability Evaluation

Condition Reports

CR-WF3-2005-03127

Work Orders

WO 69125

Section 1R12: Maintenance Rule

Procedures:

NUMBER	TITLE	REVISION
MM-006-021	Charging Pump Maintenance	4
DC-121	Maintenance Rule	1

Condition Reports

CR-WF3-2004-1501	CR-WF3-2004-1280	CR-WF3-2004-1236
CR-WF3-2005-0332	CR-WF3-2004-1274	CR-WF3-2004-1224

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
ER-W3-2004-0250	Charging Pump AB Internal Check Valves	0
Regulatory Guide 1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	

Section 1R13: Risk Assessments

Procedure

NUMBER	TITLE	REVISIONS
CEP-IST-1	IST Bases Document	3

Section 1R15: Operability Evaluations

Procedures:

NUMBER	TITLE	REVISION
CEP-IST-1	IST Bases Document	3

Condition Reports

CR-WF3-2005-3533	CR-WF3-2005-3127	CR-WF3-2005-3533
CR-WF3-2002-1539	CR-WF3-2005-1344	CR-WF3-2005-1346
CR-WF3-2005-1346		CR-WF3-2005-3577
CR-WF3-2005-3431		

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
ER-W3-2004-0615	UHS Impact Due to Increased Heat Loads Following a Design Basis Tornado Event	0

Work Orders
72259, 64591

Section 1R19: Postmaintenance Testing

Procedures:

NUMBER	TITLE	REVISION
STA-001-005	Leakage Testing of Air and Nitrogen Accumulators for Safety Related Valves	7
OP-903-008	Reactor Coolant System Isolation Leakage Test	6

Condition Reports

CR-WF3-2005-3364
CR-WF3-2002-3430
CR-WF3-2005-3857

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
Technical Manual 457001492	Anchor Darling Tilting Disc Check	1

Work Orders
70226, 26697, 55925

Section 1R20: Refueling and Outage Activities

Calculations

Calculation ECS05-003, "CEOG Tasks 555, 604, and 616 Information Evaluation for EPU," Revision 0

CE NPSD-616, Alternate Containment Closure Criterion During Loss of Shutdown Cooling at Mid-Loop Conditions," 1991

Procedures

NUMBER	TITLE	REVISION
OP-010-006	Outage Operations	0
MM-008-001	Maintenance Access Hatch and Maintenance Hatch Shield Door Opening, Inspection, and Closing	9
PLG-009-018	Containment Coordination	2
OP-901-131	Shutdown Cooling Malfunction	2

Miscellaneous

Waterford SES Unit 3 Generic Letter 87-12, "Loss of Residual Heat Removal While The Reactor Coolant System is Partially Filled," Response dated September 21, 1987

Waterford SES Unit 3 Generic Letter 88-17, "Loss of Decay Heat Removal Response to Expeditious Actions" dated December 23, 1988

Waterford SES Unit 3 Generic Letter 88-17, "Loss of Decay Heat Removal Response to Programmed Enhancements" dated February 1, 1989

Waterford 3 FSAR, Section 9.3.6.3.4, "Loss of SDCS with RCS Partially Filled

Waterford 3 RF 13 Outage Briefing/Turnover Report, dated April 22, 2005

Section 1R22: Surveillance Testing

Procedures

NUMBER	TITLE	REVISION
NE-002-002	Variable Tavg Test	12

Section 1R22: Surveillance Testing

Procedure

NUMBER	TITLE	REVISIONS
OP-903-130	Verification of Locked Valves and Breakers	2
OP-903-100	MOV Overload Bypass Test	7
CEP-IST-1	IST Bases Document	3

Condition Reports

CR-WF3-2005-3840	CR-WF3-2002-1410	CR-WF3-2005-3837
CR-WF3-2002-3924	CR-WF3-2005-3830	
CR-WF3-2005-3840	CR-WF3-2005-3858	

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
ER-W3-2003-0010	Emergency Diesel Generator (EDG) Fuel Oil Filter Tubing Modification	0
EOS 05-0052	SI Pump A Recirc Isol VLV (SI-120A) TOL	0

Work Order

51004392, 50984800, 52824, 20260, 52825, 72148, 72161

Section 1R23: Temporary Plant Modifications

Procedures:

NUMBER	TITLE	REVISION
UNT-005-004	Temporary Alteration Control	16

Condition Reports

CR-WF3-2005-3710	CR-WF3-2005-3983	CR-WF3-1999-1207

CR-WF3-1997-0552	CR-WF3-1997-2012	CR-WF3-2005-3983

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
ER-W3-2005-0433	Leak on Line 2MS2-123, Downstream of 2MS12-47B, Drip Pot Upstream of Main Steam Isolation Valve #2	0
CNRO-2005-00053	Request to Deviate from Provisions of ASME Code Case N523-2	

Work Order

72923

Section 4OA2: Identification and Resolution of Problems

Waterford Steam Electric Station, Unit 3, Amendment No. 168, "Amendment for a Previously Unreviewed Safety Question Regarding Design Basis Concerning Tornado Missile"

Condition Reports: CR-WF3-2005-3431, CR-WF3-2005-1247

Engineering Calculation EC-C97-003, Revision 1, "Probabilistic Evaluation of Tornado Missile Strike for Waterford 3 Nuclear Station

Engineering Request ER-W3-2004-0615-0000, Revision 0, "Cycle Specific Analysis for UHS Impact Due to Spent Fuel Pool Cooling

LIST OF ACRONYMS

CFR	*Code of Federal Regulations*
DAC	derived air concentration
DCT	dry cooling towers
DG	diesel generator
FOST	fuel oil storage tank
NCV	noncited violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PDR	Public Document Room
RCS	reactor coolant system
TS	technical specification
UFSAR	updated final safety analysis report