UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4005

February 7, 2005

Joseph E. Venable
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Entergy Operations, Inc.
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Killona, LA 70066-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000382/2004005

Dear Mr. Venable:

On December 31, 2004, the NRC completed an inspection at your Waterford Steam Electric Station, Unit 3. The enclosed report documents the inspection findings which were discussed on January 10, 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 has identified four issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that a violation is associated with each issue. These violations are being treated as a noncited violation, 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/

William B. Jones, Chief
Project Branch E
Division of Reactor Projects

Docket: 50-382
License: NPF-38

Enclosure:
NRC Inspection Report
050000382/2004005
w/attachment: Supplemental Information

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RIV:RI/DRP/E	SRI:DRP/E	C:DRS/EB	C:DRS/PSB	C:DRS/OB
GFLarkin	MCHay	JAClark	MPShannon	ATGody
E-WBJ	***/RA/***	***/RA/***	***/RA/***	***/RA/***
2/7/05	1/18/05	1/20/05	1/25/05	1/20/05

C:DRS/PEB	C:DRP/E		
LJSmith	WBJones		
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1/24/05	2/7/05		

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-382
License:	NPF-38
Report:	05000382/2004005
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	September 27 through December 31, 2004
Inspectors:	M. C. Hay, Senior Resident Inspector G. F. Larkin, Resident Inspector R. Mullikin, Senior Reactor Inspector, Plant Engineering Branch C. J. Paulk, Senior Reactor Inspector, Engineering Branch M.E. Murphy, Senior Operations Engineer J. M. Mateychick Reactor Inspector, Engineering Branch W. M. McNeill, Reactor Inspector, Engineering Branch B. W. Tindell, Reactor Inspector, Plant Engineering Branch D. R. Carter, Health Physicist, Plant Support Branch B. K. Tharakan, Health Physicist, Plant Support Branch P. J. Elkman, Emergency Preparedness Analyst
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Approved By:	W. B. Jones, Chief, Project Branch E
ATTACHMENT:	Supplemental Information

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000382/2004-005; 09/27/2004-12/31/2004; Waterford Steam Electric Station, Unit 3; Surveillance Testing, Access Control to Radiological Significant Areas, Problem Identification and Resolution, Inspection Followup

The report covered a 14-week period of inspection by resident inspectors, regional reactor engineering inspectors, a regional health physicist, and a regional emergency preparedness inspector. The inspectors identified four Green findings. The significance of most findings are indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." 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 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

• Green. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to implement effective corrective actions to prevent recurrence for a significant condition adverse to quality affecting operability of the main feedwater isolation valves. Specifically, on multiple occasions accumulator over-pressure conditions have occurred, resulting from degraded hydraulic fluid adversely affecting the hydraulic actuator pressure relief system. These over-pressure conditions potentially result in valve closure stroke times outside design basis values.

The finding was greater than minor because it is associated with the mitigating systems cornerstone objective to ensure the capability of systems that respond to initiating events to prevent undesirable consequences. The finding was evaluated using the Inspection Manual Chapter 0609, Significance Determination Process, Phase 1 Worksheet for mitigating systems. The finding was determined to be of very low risk significance because the over-pressure conditions did not represent an actual loss of a safety function of a single train for greater than its Technical Specification allowed outage time (Section 4OA2).

C Green. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix R, Section III.L.3, for the failure to provide electrical independence in the Waterford design that included a neutral (ground) wire that was not isolated from the control room during transfer to the alternative shutdown panel. Entergy initiated Condition Report WF3-2004-03541 to track the modification to isolate the neutral wire for the affected safe shutdown circuits. The modification will bring Waterford into compliance with Appendix R.

This finding is greater than minor because it was associated with the mitigating systems cornerstone attribute of protection against external factors (fire) and it has the potential to impact the mitigating systems cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. The violation is associated with degradation of a fire protection feature. Using Part 1 of the Inspection Manual Chapter 0609, fire protection Significance Determination Process Phase 1 Worksheet, the performance issue was determined to be in the postfire safe shutdown category. The degradation rating was low based on Entergy's determination that there were no existing conditions that would prevent the plant from achieving and maintaining a safe shutdown in the event of a control room fire, if the installed protective devices always operated within their designed tripping characteristics. Therefore, the finding screens as Green or of very low safety significance in the Phase 1 Worksheet. This violation is being treated as a noncited violation consistent with Section VI.A of the Enforcement Policy (Section 4OA5).

Cornerstone: Barrier Integrity

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the failure to establish adequate test controls for leak testing those portions of fluid systems outside containment that could contain highly radioactive fluid during a serious transient or accident. This performance deficiency could result in underestimating the leak rate of highly radioactive fluid into the reactor auxiliary building during accident conditions.

  The finding was 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 the Inspection Manual Chapter 0609, Significance Determination Process, Phase 1 Worksheet for barrier integrity. The finding was only of very low safety significance because it did not represent an actual reduction of the atmospheric pressure control function of the reactor containment and it did not result in an actual open pathway affecting the physical integrity of reactor containment (Section 1R22).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a self-revealing noncited violation of Technical Specification 6.8.1 because Entergy failed to follow radiation work permit requirements. On November 12, 2003, two individuals' faces became contaminated while performing maintenance on Steam Generator 2 manway studs. Personnel contamination monitors alarmed upon the exit of the individuals from the controlled access area. These alarms prompted Entergy to investigate the events and conclude that multiple violations of Radiation Work Permit 2003-1509, Task 3, occurred. Specifically, workers did not: (1) wear face shields or power visors during stud work, (2) have constant radiation protection technician coverage, (3) wear telemetry electronic dosimeters and move them to the head, or (4) wear lapel air samplers. This finding had human performance crosscutting aspects.

This finding is greater than minor because it is associated with the Occupational Radiation Safety attribute of exposure control and affected the cornerstone objective because not following radiation work permit requirements could increase personnel dose. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that the finding was of very low safety significance because it did not involve: (1) as low as is reasonably achievable planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding was entered into Entergy's corrective action program (Section 2OS1).

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status: The plant was operated at approximately 100 percent power from September 27 through December 31, 2004, except when reactor power was reduced to approximately 95 percent on October 22, 2004, to conduct moderator temperature coefficient testing and to approximately 88 percent on November 18, 2004, to conduct high-pressure turbine valve testing.

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

1R02 Evaluation of Changes, Tests, or Experiments (71111.02)

a. Inspection Scope

The inspection procedure requires a minimum sample size of 6 evaluations and 12 screenings. The inspectors reviewed 7 licensee-performed safety evaluations to verify that Entergy Operations, Inc. (Entergy) had appropriately considered the conditions under which Entergy may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval. Entergy performed these evaluations since the last NRC inspection of activities performed by Entergy personnel pursuant to 10 CFR 50.59, “Changes, Tests and Experiments.”

The inspectors reviewed 11 licensee-performed screenings in which a full evaluation had been excluded. The inspectors did such to ensure consistency with the requirements of 10 CFR 50.59 in the exclusion of a full evaluation. The inspectors also reviewed 18 applicability determinations in which licensee personnel excluded screenings to ensure consistency with the requirements of 10 CFR 50.59 in the exclusion of a screening and/or evaluation.

The inspectors reviewed a sample of 6 of the 67 corrective action documents written by Entergy since the last NRC inspection involving safety evaluation-related activities to determine whether Entergy properly identified and subsequently resolved problems and/or deficiencies.

b. Findings

An Entergy-identified noncited violation is documented in Section 4OA7 of this report.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial System Walkdowns

The inspectors performed the following three partial system equipment alignment inspections during this inspection period:

- On December 2, 2004, the inspectors performed a partial equipment alignment inspection of low-pressure safety injection (LPSI) system Train A while LPSI system Train B was inoperable. The inspectors performed a review of select maintenance work orders and corrective action documents to assess the material condition and performance of LPSI system Train A. A walkdown of accessible portions of the system was performed to assess material condition, such as system leaks and housekeeping issues, that could adversely affect system operability. System configuration was assessed using Operating Procedure OP-009-008, "Safety Injection System," Revision 16, as well as applicable sections of the Updated Final Safety Analysis Report.

- On December 7, 2004, the inspectors performed a partial equipment alignment inspection of emergency diesel generating system Train B while emergent repairs were being performed on the other redundant train. The inspectors performed a walkdown of accessible portions of the system assessing material condition, housekeeping issues, and system configuration. System configuration was assessed using Operating Procedure OP-009-002, “Emergency Diesel Generator,” Revision 18.

- On November 18, 2004, the inspectors performed a partial equipment alignment inspection of offsite electrical power supply Train A during planned maintenance activities being performed on the other redundant train. The inspectors performed a walkdown of accessible portions of the electrical distribution system assessing material condition, housekeeping issues, and system configuration. System configuration was assessed using Operating Procedure OP-006-001, “Plant Distribution (7KV, 4KV & SSD) Systems,” Revision 12.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors conducted six inspections to assess whether Entergy 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 of the reactor auxiliary building were inspected:

- Fire Zones 2, 16, 17, 18, 19, 20, 21, and 33 on October 10, 2004

- Fire Zones 1A, 5, 6, 7, 8A, 8B, and 8C on November 10, 2004

- Fire Zones 1A, 1B, 1C, 2, and 3 on November 12, 2004

- Fire Zones 1A, 8A, 8B, 8C, and 15 on November 22, 2004

- Fire Zones 1A, 15, 16,17, 18, 19, 20, 21, and 23 on December 16, 2004

- Fire Zones 2, Wet and Dry Cooling Tower Train B, Fire Pump House, Roof East, and Roof West on December 21, 2004

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors performed a semiannual inspection of internal flood protection features in the turbine generator building switchgear room. The swichgear room contains portions of both physically independent electrical circuits between the offsite transmission network and the onsite Class 1E distribution system. The inspection included a review of the Updated Final Safety Analysis Report (UFSAR), selected design calculations, Regulatory Guide 1.102, "Flood Protection for Nuclear Plants;" Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," and a walkdown of flood protection features in the turbine generator building switchgear room.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11, 71111.11B)

.1 Licensed Operator Requalification

a. Inspection Scope

On October 5, 2004, the inspectors observed a licensed operator simulator training scenario. During the scenario, operators responded to problems associated with the main transformer, emergency diesel generators, a loss of off-site power, and a station blackout with a concurrent loss of the emergency feedwater turbine-driven Pump A/B. 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

- 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.

.2 Biennial Inspection

c. Inspection Scope

The inspector reviewed the annual operating examination test results for 2004. Since this was the first half of the biennial requalification cycle, the licensee had not yet administered the written examination. These results were assessed to determine if they were consistent with NUREG 1021, “Operator Licensing Examination Standards for Power Reactors,” Revision 8, Supplement 1, guidance and Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process,” requirements. This review included examination of test results, which included no crew or individual failures out of a total of 48 licensed operators during the scenario examinations. There were 4 failures out of 48 licensed operators during the job performance measure examinations, all 4 operators were remediated and re-examined successfully prior to their return to licensed duties.

d. 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 Entergy met the requirements of 10 CFR 50.65(a)(4) for assessing and managing any increase in risk from these activities. The following three risk evaluations were reviewed:

- On October 7, 2004, during emergent maintenance on the emergency diesel generator Train B fuel oil transfer pump control switch

- On October 18-21, 2004, during planned maintenance on the Waterford Steam Electric Station (Waterford 3) switchyard west bus

- On December 16-23, 2004, during emergent maintenance on the Reactor Protection System, Trip Path 1

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the technical adequacy of two 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 Technical Specification, UFSAR, and design-bases requirements and that the potential risk increase contributed by the degraded equipment was thoroughly evaluated. The following evaluations were reviewed:

- Operability evaluation addressing inadequate ASME Section XI pressure testing on safety injection (SI) recirculation suction piping (Condition Report CR-WF3-2004-03454)

- Operability evaluation addressing the chilled water system environmental qualifications due to the radiation shine from the controlled ventilation area system (CVAS) filter trains following an accident condition (Condition Report CR-WF3-2004-3560)

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The procedure requires the review of a minimum of 5 permanent plant modifications. The inspectors reviewed 12 permanent plant modification packages and associated documentation, such as 10 CFR 50.59 review screens and safety evaluations, to verify that they were performed in accordance with plant procedures. The inspectors also reviewed the procedures governing plant modifications to evaluate the effectiveness of the programs for implementing modifications to risk-significant systems, structures, and components, such that these changes did not adversely affect the design and licensing basis of the facility.

The inspectors interviewed the cognizant design and system engineers for the identified modifications as to their understanding of the modification packages.

The inspectors evaluated the effectiveness of Entergy's corrective action process to identify and correct problems concerning the performance of permanent plant modifications. In this effort, the inspectors reviewed 13 corrective action documents and the subsequent corrective actions pertaining to licensee-identified problems and errors in the performance of permanent plant modifications.

b. 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, Technical Specifications, and Entergy's procedural requirements. The inspectors reviewed the testing results for the following two components:

- Charging Pump A, following emergent repairs for packing seal replacement on November 23, 2004

- Main feedwater isolation Valve (MFIV) 1, following emergent repairs on the valve's hydraulic fluid system on December 21, 2004

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed or reviewed the following three 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 surveillance tests met Technical Specifications, the UFSAR, and Entergy's procedural requirements:

- Surveillance Procedure OP-903-024, "Reactor Coolant System Water Inventory Balance," Revision 13, performed on December 6, 2004. This surveillance determines the quantity of identified and unidentified leakage from the reactor coolant system during plant steady state operations.

- Surveillance Procedure OP-903-118, "Primary Auxiliaries Quarterly IST Valve Tests," Revision 6, performed on December 17, 2004. This surveillance verified the functional capability for containment atmospheric purge Valves CAP-103 and CAP-104 to close within required stroke times.

- Surveillance Procedure OP-903-110, “RAB Fluid Systems Leak Test,” Revision 13, performed on September 9, 2004. This surveillance verified the leak tightness of fluid systems located outside of containment that could contain highly radioactive fluid during a serious transient or accident.

b. Findings

Introduction. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the failure to establish adequate test controls for leak testing those portions of fluid systems outside containment that could contain highly radioactive fluid during a serious transient or accident. This performance deficiency could result in underestimating the leak rate of highly radioactive fluid into the reactor auxiliary building during accident conditions.

Description. On September 9, 2004, the inspectors observed surveillance Test OP-903-110, “RAB Fluid Systems Leak Test” on the SI system sump Train B outlet piping between SI Valves SI-602B and SI-604B. Valve SI-602B is the outside containment isolation valve designed to isolate containment from the reactor water storage pool (RWSP), high-pressure safety injection system pump Train B, low pressure safety injection (LPSI) pump Train B, and containment spray (CS) pump Train B prior to a recirculation actuation signal. Valve SI-604B is located between Valve SI-602B and the RWSP. Valve SI-604B is a check valve that prevents back flow from the RWSP into the safety injection sump. The purpose of surveillance Test OP-903-110 was to determine the leak rate of systems outside containment that contain highly contaminated fluids during a serious transient or accident. This contaminated fluid could leak into that portion of the reactor auxiliary building filtered by the CVAS. The CVAS system is designed to remove airborne contamination to acceptable levels prior to being released outside the plant. The inspectors noted that the radiological dose analysis following accident conditions was based on not exceeding a one gallon per minute leak rate from highly contaminated systems outside containment.

The inspectors noted that the peak containment pressure as stated in the UFSAR was 44 psig during a loss of coolant accident (LOCA) and that surveillance Procedure OP-903-110 only pressurized the piping section from Valves SI-602B to SI-604B to 11 psig using instrument air during the surveillance. The inspectors also noted that surveillance Procedure OP-903-110 did not provide a means to correlate air leakage at 11 psig test pressure to 44 psig design pressure. The inspectors were concerned that leakage testing performed at the reduced pressures may not reflect actual leakage during design basis accident conditions.

The inspectors discussed these observations with Entergy, which also concluded the testing methodology was inappropriate. During Entergy’s review, additional leakage testing discrepancies were identified and appropriately entered into Entergy’s corrective action process for resolution.

Analysis. The deficiency associated with this finding was the failure to develop an adequate test program to identify system leakage in potentially highly contaminated systems. The finding was 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 [SDP], Appendix A, SDP Phase 1 Screening Worksheet, dated December 1, 2004, for Initiating Events, Mitigating Systems, and Barrier Cornerstones. The finding was determined to be of very low safety significance because it did not represent an actual reduction of the atmospheric pressure control function of the reactor containment and it did not result in an actual open pathway affecting the physical integrity of reactor containment.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," states, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service. The failure to establish adequate testing controls to ensure a highly contaminated piping system outside containment would perform satisfactorily in service is a violation of 10 CFR Part 50, Appendix B, Criterion XI. Because the failure to establish adequate testing controls was of very low safety significance and has been entered into Entergy's corrective action program as Condition Reports 2004-3454, 2004-3457, 2004-3888, and 2004-4048, this violation is being treated as a noncited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-382/2004005-01, Inadequate Test Controls to Identify Leakage of Potentially Highly Radioactive Fluids Outside Containment.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Temporary Shielding Request 2004-0047, "Protect Essential Chillers A and B from Radiation Degradation." This plant modification was installed to reduce post-LOCA radiation to the essential chillers from the CVAS filters. The inspectors reviewed the safety screening, design documents, UFSAR, and applicable Technical Specifications 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. The inspectors reviewed the adequacy of postinstallation tests and test results to confirm that the actual impact of the temporary modification on the permanent system and interfacing systems was adequately verified.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert Notification System Testing (71114.02)

a. Inspection Scope

The inspector discussed with Entergy the status of offsite siren and other public notification systems to determine the adequacy of Entergy's methods for testing the alert and notification system in accordance with 10 CFR Part 50, Appendix E and reviewed the addition of two additional offsite sirens in January 2004. Entergy's alert and notification system testing program was compared with criteria in:

- NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1

- Federal Emergency Management Agency Report REP-10, "Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plants"

- Entergy's updated Federal Emergency Management Agency approved alert and notification system design report, dated October 2004

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing (71114.03)

a. Inspection Scope

The inspectors discussed with Entergy the status of primary and backup systems for staffing emergency response facilities during an emergency, including Entergy's migration to an updated computer-based notification system. The inspector reviewed Procedure EP-002-010, "Notifications and Communications," Revision 29, and Procedure EP-002-015, "Emergency Responder Activation," Revision 8, to determine Entergy's ability to staff emergency response facilities in accordance with the Entergy emergency plan and the requirements of 10 CFR Part 50, Appendix E. The inspectors also compared the results of 21 notification pager drills to emergency response facility activation requirements to determine the performance of Entergy's emergency response organization augmentation system.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors performed an onsite review of Revision 30, Change 1, to the Waterford Steam Electric Station, Unit 3, Emergency Plan. This revision added details to the emergency plan regarding shelter-in-place as a protective action recommendation to offsite authorities. To determine if the revision decreased the effectiveness of the emergency plan it was compared to:

- Waterford Steam Electric Station, Unit 3, Emergency Plan, Revision 29,
- Criteria of NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Revision 1,
- Criteria of Procedure EP-305, “10CFR50.54(q) Review Program,” Revision 1,
- Criteria of Desk Guide 09, “Emergency Plan and Procedure Maintenance, Revisions and Changes Guidelines,” Revision 3, and
- Requirements of 10 CFR 50.47(b) and 50.54(q).

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

a. Inspection Scope

The inspectors reviewed the following documents related to Entergy’s corrective action program to determine Entergy’s ability to identify and correct problems in accordance with 10 CFR 50.47(b)(14) and 10 CFR Part 50, Appendix E.

- Procedure EN-LI-102, “Corrective Action Process,” Revision 0
- Desk Guide 17, “Drill Control Team Documentation,” Revision 0
- W3F3-2003-0016, Emergency Preparedness Audit, April/May 2003
- Quality Assurance Audit Report QA-7-2004-WF3-1
- Summaries of 212 corrective actions assigned to the emergency preparedness department during calendar years 2003 and 2004
- Details of 20 selected condition reports

The inspector also observed the emergency operations facility during one tabletop scenario and the technical support center during one site-wide drill to evaluate the effectiveness of completed corrective actions.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

On September 30, 2004, the inspectors reviewed the drill scenario and observed activities in the simulated control room and the Emergency Operations Facility. The drill scenario simulated Mississippi River flooding conditions, equipment failures, site evacuation, a reactor core transient with leakage of reactor coolant, and the release of radioactive material offsite. The inspectors evaluated performance by focusing on the risk significant activities of emergency classification, notification, and protective action recommendations. In addition, the inspectors reviewed the drill critiques and the resolution of identified performance weaknesses.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiological Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess Entergy's performance in implementing physical and administrative controls, including worker adherence to these controls for airborne radioactivity areas, radiation areas, high radiation areas, locked high radiation areas, and very high radiation areas. The inspectors used the requirements in 10 CFR Part 20, the Technical Specifications, and Entergy's procedures required by the Technical Specifications as criteria for determining compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation packages reported by Entergy in the Occupational Radiation Safety Cornerstone

- Controls (surveys, posting, and barricades) of three radiation, high radiation, or airborne radioactivity areas
- Radiation work permit procedure, engineering controls, and air sampler locations
- Conformity of electronic personal dosimeter alarm setpoints with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms
- Barrier integrity and performance of engineering controls in one potential airborne radioactivity work area
- Physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within the spent fuel storage pool
- Self-assessments and audits related to the access control program since the last inspection
- Corrective action documents related to access controls
- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies
- Radiation work permit briefings and worker instructions
- Adequacy of radiological controls such as required surveys, radiation protection job coverage, and contamination controls during job performance
- Dosimetry placement in high radiation work areas with significant dose rate gradients
- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas
- Controls for special areas that have the potential to become very high radiation areas during certain plant operations
- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas
- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

Either because the conditions did not exist or an event had not occurred, no opportunities were available to review the following items:

- Adequacy of Entergy's internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent

- Licensee event reports and special reports related to the access control program since the last inspection

The inspectors completed 21 of the required 21 samples.

b. Findings

Introduction. The inspectors identified a Green, self-revealing, noncited violation of Technical Specification 6.8.1 for failure to follow radiation work permit requirements.

Description. On November 12, 2003, two individuals' faces became contaminated with approximately 150 corrected counts per minute of radioactivity while performing maintenance on Steam Generator 2 manway studs. The inspectors reviewed Condition Report CR-2003-3583 that was written to address the issue. The individuals were working under Task 3 of Radiation Work Permit 2003-1509. Upon exiting the controlled access area, the workers alarmed the personal contamination monitors. These alarms prompted Entergy to investigate the events and conclude that multiple violations of the radiation work permit had occurred. Specifically, the following radiation work permit requirements were not followed:

- Workers did not wear face shields or power visors during stud work

- Radiation protection did not ensure that a breathing zone air sample was taken

- Continuous radiation protection job coverage or direct communication by remote camera were not provided

- Workers were not wearing telemetry electronic dosimetry, nor were the electronic dosimeters moved to the workers' heads as required by the radiation work permit

Entergy performed whole body counts on the two individuals. The individuals were not assigned additional exposure due to the facial contaminations. Entergy also performed air samples of the area and assessed if additional worker dose was received due to not moving dosimetry or for not performing continuous radiation protection job coverage. No additional exposure was received by the workers.

Analysis. The failure to follow radiation work permit requirements is a performance deficiency. This finding is greater than minor because it was associated with a cornerstone attribute (exposure control) and affected the associated cornerstone objective to ensure the adequate protection of worker health and safety from exposure to radiation from radioactive material, because not following radiation work permit requirements could increase personnel dose. The finding involved workers' unplanned, unintended dose or potential for such a dose that could have been significantly greater as a result of a single, minor, reasonable alteration of the circumstances. When processed through the Occupational Radiation Safety SDP, the finding was of very low safety significance because it did not involve: (1) as low as is reasonably achievable (ALARA) planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding had crosscutting aspects associated with human performance. When licensee personnel failed to follow radiation work permit instructions, their actions directly contributed to the finding.

Enforcement. Technical Specification 6.8.1 requires written procedures be established, implemented, and maintained covering the activities recommended in Appendix A of Regulatory Guide 1.33, Revision 2. Regulatory Guide 1.33, Appendix A, Section 7e, requires procedures for access control to radiation areas, including a radiation work permit system. Radiation Work Permit 2003-1509, Task 3, required, in part, workers wear face shields or power visors during stud work, have constant radiation protection technician coverage, wear telemetry electronic dosimeters and move them to the head, and perform an air sample. The failure to comply with radiation work permit requirements is a violation of Technical Specifications.

Because the failure to follow radiation work permit requirements was determined to be of very low safety significance and has been entered into Entergy's corrective action program as Condition Report CR-2003-3583, this violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy:
NCV 50-382/2004005-02, Technical Specification Violation for Failure to Follow Radiation Work Permit Requirements.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector assessed licensee performance with respect to maintaining individual and collective radiation exposures ALARA. The inspector used the requirements in 10 CFR Part 20 and Entergy's procedures required by Technical Specifications as criteria for determining compliance. The inspector interviewed Entergy personnel and reviewed:

- Current 3-year rolling average collective exposure
- Site-specific trends in collective exposures, plant historical data, and source-term measurements
- Site-specific ALARA procedures
- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding
- First-line job supervisors' contribution to ensuring work activities are conducted in a dose efficient manner

- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry

- Source-term control strategy or justifications for not pursuing such exposure reduction initiatives

- Specific sources identified by Entergy for exposure reduction actions and priorities established for these actions and results achieved since the last refueling cycle

- Declared pregnant workers during the current assessment period, monitoring controls, and the exposure results

- Self-assessments, audits, and special reports related to the ALARA program since the last inspection

- Corrective action documents related to the ALARA program and followup activities, such as initial problem identification, characterization, and tracking

- Effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies

The inspector completed 8 of the required 15 samples and 4 of the optional samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors sampled Entergy's submittals for the performance indicators listed below for the period from April 2003 through September 2004. To verify the accuracy of the performance indicator data reported during that period, performance indicator definitions and guidance contained in NEI (Nuclear Energy Institute) 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element. Entergy's performance indicator data were also reviewed against the requirements of Procedure EN-EP-201, "Emergency Planning Performance Indicators," Revision 1, and Desk Guide 15, "Performance Indicators," Revision 1.

Initiating Events Cornerstone

- Unplanned Scrams per 7,000 Critical Hours

No reactor scrams have occurred in the past four quarters. Indicator value remains at 0.0.

Emergency Preparedness Cornerstone

- Drill and Exercise Performance
- Emergency Response Organization Participation
- Alert and Notification System Reliability

The inspectors reviewed a 100 percent sample of drill and exercise scenarios, licensed operator simulator training sessions, notification forms, and attendance and critique records associated with training sessions, drills, and exercises conducted during the verification period. The inspectors reviewed selected emergency responder qualification, training, and drill participation records. The inspector reviewed a 100 percent sample of siren and helicopter loudspeaker test records and reviewed siren maintenance records and procedures. The inspectors also interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness Performance Indicator

Entergy's records reviewed included corrective action program records for Technical Specification-required locked high radiation areas, very high radiation areas as defined in 10 CFR 20.1003, and unplanned exposure occurrences from March 2003 to confirm that any occurrences were properly recorded as performance indicators as defined in NEI 99-02. Controlled access area exits with exposures greater than 100 millirems were reviewed and selected examples were examined to determine whether they were within the dose projections of the governing radiation exposure permits. The inspectors interviewed Entergy personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspectors toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled.

Public Radiation Safety Cornerstone

- Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences Performance Indicator

Licensee records reviewed included radiological effluent release program corrective action records and annual effluent release reports documented since March 2003 to determine if any liquid or gaseous effluent releases resulted in events that exceeded the performance indicator thresholds. The inspectors interviewed Entergy personnel that were accountable for collecting and evaluating the performance indicator data.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Annual Sample Review

a. Inspection Scope

The inspectors reviewed performance and facility problems associated with the emergency preparedness program documented in Entergy's corrective action program, audits, and drill reports during calendar years 2003 and 2004. The inspectors selected 15 items to verify effective corrective action through observation of tabletop and facility drills and direct inspection.

The inspectors also assessed implementation of Entergy's corrective action process involving multiple actuator over-pressure conditions affecting operability of the main feedwater isolation valves.

b. Findings and Observations

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to implement effective corrective actions to prevent recurrence for a significant condition adverse to quality affecting operability of the main feedwater isolation valves. Specifically, on multiple occasions accumulator over-pressure conditions have occurred resulting from degraded hydraulic fluid adversely affecting the pressure relief system. These over-pressure conditions would potentially result in valve closure stroke times outside design basis values.

Description: The inspectors reviewed Condition Report CR-WF3-2004-4093 pertaining to the failure of MFIV Train A, Accumulator A, thermal hydraulic relief system resulting in an over-pressure condition on December 16, 2004. The MFIV FW-184A and FW-184B automatically close for containment isolation and to limit reactor coolant system cooldown during certain design basis accidents. Acceptable valve closure times range from 2.3 to 5.0 seconds. The lower closure limit is to preclude a water-hammer, which can be induced by fast valve closure. Each MFIV hydraulic operating system is equipped with two hydraulic accumulators. Both accumulators are required for proper valve operation. In order to limit valve speed, accumulator pressure is limited to 5900 psig. Each accumulator is equipped with a thermal relief valve to prevent over-pressurization.

On December 16, 2004, a white trouble light illuminated in the control room for MFIV Train A, Accumulator A. Entergy determined the accumulator pressure was 5906 psig due to failure of the hydraulic accumulator thermal relief valve to relieve pressure at its 5800 psig setpoint. Subsequently, operators declared the MFIV inoperable. Attempts to relieve the pressure by opening the thermal relief bypass valve were unsuccessful.

Entergy's failure modes analysis team determined that a flow restricting orifice upstream of the thermal relief valve and the thermal relief bypass valve was obstructed with particulate. The particulate was black and identified as gelled fryquel. Gelled fryquel has contributed numerous times to past failures of the MFIV pressure relief system, affecting both trains. The inspectors noted that over-pressure conditions were experienced on MFIV Train B, Accumulator B, on March 29, 2004, and on MFIV Train A, Accumulator B, on July 20, 2004. The inspectors also noted that additional examples were discussed in NRC Inspection Report 05000382/2004006, Section 4OA2 e. The inspectors determined Entergy's corrective actions to prevent the over-pressure condition from affecting the operability of the MFIV's have not been effective and have repeatedly failed to prevent recurrence of this significant condition adverse to quality.

Analysis. The deficiency associated with this finding was the failure to establish corrective measures to prevent recurrence of a significant condition adverse to quality. Specifically, corrective actions established to address over-pressure hydraulic fluid conditions were not effectively implemented and failed to prevent recurrence resulting in the MFIV's being declared inoperable. The inspectors determined that the issue was more than minor in significance because it affected the mitigating systems cornerstone objective to ensure the availability of systems that respond to initiating events. The inspectors utilized the Inspection Manual Chapter 0609, Significance Determination Process [SDP], Appendix A, SDP Phase 1 Screening Worksheet, dated December 1, 2004, for Initiating Events, Mitigating Systems, and Barrier Cornerstones to assess the safety significance. The finding was determined to be of very low risk significance because, in each over-pressure condition identified, the affected train was inoperable for less than the Technical Specification allowed outage time.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires in part, that measures be established to assure that conditions adverse to quality 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 establish corrective measures to prevent recurrence of main feedwater isolation valve over-pressure conditions is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. Because the failure to prevent recurrence was of very low safety significance and has been entered into Entergy's corrective action program as Condition Report CR-WF3-2004-4093, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: (NCV 05000382/2004005-03) Failure to Prevent a Reoccurrence of an Over-Pressure Condition in Main Feed Water Isolation Valve Hydraulic Operating Systems.

.2 Semiannual Trend Review

a. Inspection Scope

On December 24, 2004, the inspectors completed the semiannual review of Entergy's identified trends for evidence that other significant safety issues may exist. The inspectors' review focused on repetitive equipment issues, but also considered the results of screening the corrective action program, self-assessment reports, control room logs, quality assurance audits, and department self-assessments to determine if additional adverse trends existed. The inspectors compared and contrasted their results with the results contained in Entergy's latest quarterly trend reports. For those areas where trends were documented in the corrective action program, the inspectors verified that Entergy had corrective actions planned or in place to address the trend. The inspectors also evaluated the corrective actions against Entergy's procedural requirements of Procedure LI-102, "Corrective Action Program." The inspectors' review nominally considered the 6-month period of July through December 2004.

b. Findings and Observations

No findings of significance were identified. The inspectors concluded that, in general, Entergy had adequately identified trends in areas within the scope of this inspection.

.3 Effectiveness of Problem Identification and Resolution Processes

Section 2OS2 evaluated the effectiveness of Entergy's problem identification and resolution processes regarding exposure tracking, higher than planned exposure levels, and radiation worker practices. The inspectors reviewed the corrective action documents listed in the attachment against Entergy's problem identification and resolution program requirements. No findings of significance were identified.

4OA4 Crosscutting Aspects of Findings

Section 2OS1 described an NCV with human performance crosscutting aspects, which involved a failure to follow radiation work permit requirements.

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000382/2000007-04): Alternate Shutdown Panel May Not be Electrically Isolated from a Control Room Fire Due to Multiple Spurious Actuations

a. Inspection Scope

The inspectors previously identified an unresolved item that certain safe shutdown circuits were not electrically independent from a fire in the control room. This issue was made unresolved pending further NRC review of the vulnerability of Entergy's circuit design. During this inspection, the inspectors reviewed licensee correspondence dated July 26, 2001, February 25, 2003, and July 24, 2003, and Condition Report W3-2004-03541. The inspectors also considered a preliminary evaluation of Entergy's position that was prepared by the Office of Nuclear Reactor Regulation's staff.

b. Inspection Finding

Introduction. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix R, Section III.L.3, because the alternate shutdown capability at the alternate shutdown panel was not fully independent from the control room. In the Waterford 3 design, a neutral (ground) wire was not isolated from the control room during transfer to the alternative shutdown panel.

Discussion. Entergy recognized that the Waterford 3 design included a neutral (ground) wire that was not isolated from the control room during transfer to the alternative shutdown panel. Entergy reviewed the design and concluded that the existing design did not create any conditions adverse to safe shutdown. The requirement of 10 CFR Part 50, Appendix R, Section III.L.3, states that alternate shutdown capability shall be independent from the specific fire area, in this case the control room. The inspectors reviewed Waterford's licensing basis and did not identify any approved exemptions related to this requirement.

Subsequently, Entergy submitted on July 26, 2001, a request for a deviation from Appendix R requirements for control circuit isolation between the control room and the remote shutdown panel. The basis for the request was an engineering evaluation performed by Entergy in 1997 in response to NRC Information Notice 97-01, "Improper Electrical Grounding Results in Simultaneous Fires in the Control Room and the Safe-Shutdown Equipment Room." Entergy reviewed the possible power sources that are available in the safe shutdown associated panels in the control room, cable vault, and alternate shutdown panel room and calculated the fault current available from these power sources. The available fault current value was compared to the tripping characteristics of the largest fuse or breaker in the circuit supplying the short circuit current. Entergy's analysis showed that in all analyzed worst-cases the fuse or breaker protecting the circuit will trip before the safe shutdown conductor reaches its melting point, presuming the protective devices (fuses and breakers) functioned as designed.

The NRC staff reviewed Entergy's deviation request of July 26, 2001, and tentatively agreed with Entergy's conclusions. However, the NRC staff informed Entergy that the appropriate action would be for Entergy to submit a license amendment request in lieu of a deviation request. In a letter to the NRC dated July 24, 2003, Entergy withdrew their Appendix R deviation request. Subsequently, Entergy's management decided that a modification to isolate the neutral wire for the affected safe shutdown circuits was more appropriate than submitting a license amendment request.

Analysis. This finding is greater than minor because it was associated with the mitigating systems cornerstone attribute of protection against external factors (fire) and it has the potential to impact the mitigating systems cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. The violation is associated with degradation of a fire protection feature. Using Part 1 of the Fire Protection SDP Phase 1 Worksheet in Manual Chapter 0609, Significance Determination Process [SDP], the performance issue was determined to be in the post-fire safe shutdown category. The degradation rating was low based on

Entergy's determination that there were no existing conditions that would prevent the plant from achieving and maintaining a safe shutdown in the event of a control room fire, if the installed protective devices (fuses and breakers) always operated within their designed tripping characteristics. Therefore, the finding screens as Green or of very low safety significance in the Phase 1 Worksheet.

Enforcement. Waterford 3 design included a neutral (ground) wire that was not isolated from the control room during transfer to the alternative shutdown panel. The failure to provide electrical independence, as required by 10 CFR Part 50, Appendix R, Section III.L.3, constitutes a violation of NRC requirements. Entergy initiated Condition Report WF3-2004-03541 to track the modification to isolate the neutral wire for the affected safe shutdown circuits. The modification will bring Waterford 3 into compliance with Appendix R. This violation is being treated as NCV 5000382/2004005-04, Failure to Isolate Neutral Wire during Transfer to the Alternative Shutdown Panel per 10 CFR Part 50, Appendix R, Section III.L.3, consistent with Section VI.A of the Enforcement Policy.

.2 (Closed) URI 05000382/2004004-04, Instrument Uncertainties for Auxiliary Component Cooling Water System

The inspectors evaluated flow balance test acceptance criteria for the component cooling water (CCW)/auxiliary component cooling water (ACCW) flow balance test and ER-W3-97-0174-00-00, "CCW &ACCW Flow Balance Test Acceptance Criteria." For the ACCW system flow balance test, it appeared that there was insufficient margin above the design basis minimum flows to directly accommodate instrument uncertainties.

Entergy personnel confirmed that instrument uncertainty must be considered in the parameters of the plant. This consideration of uncertainty includes either the explicit (direct) or implicit (analytical) application of instrument uncertainty as it relates to the safety function, in this case removal of heat to the ultimate heat sink.

Entergy prepared ER-W3-2004-0506-000, "Consideration of Instrument Uncertainty in the CCW - ACCW Flow Balance," to illustrate the consideration of instrument uncertainty in PE-004-024, "ACCW [auxiliary component cooling water] and CCW [component cooling water] Flow Balance." Entergy concluded that when the excess thermal capacity in the CCW/ACCW heat exchanger was considered, the necessary ACCW flow margin was implicitly available.

The inspectors independently confirmed these conclusions through calculations. This URI is closed.

.3 (Closed URI 05000382/2004004-06): Review Safety Significance of Safety Injection Valve SI-602B Leakage

This unresolved item is closed. The basis is provided in Section 4OA7.4 of this report.

4OA6 Meetings

Exit Meeting Summary

The inspectors conducted several exit meetings during the inspection period. The inspectors asked Entergy whether any materials examined during the inspection should be considered proprietary. Any proprietary information was reviewed by the inspectors and left with Entergy at the end of the inspection and no proprietary information is contained in this report. The following exit meetings were conducted:

- The inspectors presented the inspection results to J. Venable, Vice President, Operations and Station Director, and other members of licensee management on September 3, 2004. Licensee management acknowledged the inspection findings. On September 9, 2004, a telephonic conference call was conducted with the NRC staff and representatives for the licensee to clarify the issues in regard to instrument uncertainties and the opportunity for the licensee to provide additional information for review to support their position.

- On October 22, 2004, the inspectors presented the results of the inspection to Mr. J. Venable, Vice President, Operations, and other members of Entergy's management who acknowledged the findings.

- On November 11, 2004, the inspectors presented the results of the inspection to Mr. J. Venable, Vice President, Operations, and other members of licensee management who acknowledged the findings.

- On November 18, 2004, the inspector presented the inspection results to Mr. J. Venable, and other members of his staff, who acknowledged the findings.

- On December 2, 2004, the inspector presented the inspection results to Mr. J. Venable, Site Vice President, and other members of his staff who acknowledged the findings.

- On January 10, 2005, the resident inspectors presented the inspection results to Mr. J. Venable, Vice President, Operations and other members of Entergy's management at the conclusion of the inspection. Entergy acknowledged the findings presented.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by Entergy and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

.1 10 CFR 50.59 Safety Evaluations and Screening

Entergy issued Condition Report CR-WF3-2004-02861, which identified that the Safety Review Committee had found problems with safety evaluation screening activities. The Safety Review Committee (i.e., the off-site review committee) found in their review of safety evaluation screenings that, contrary to Criterion V, of Appendix B to 10 CFR Part 50, the plant staff failed to follow Plant Implementing Procedure ENS-LI-101, “10 CFR 50.59 Review Program,” Revision 4, regarding several items. Specifically, the condition report cited examples of safety evaluation screenings of procedure changes with:

- Inadequate description of the change,
- Inadequate bases for the conclusions (two examples),
- Inadequate licensing basis document search (two examples),
- Missing signatures (two examples), and
- Typographical errors.

The condition report states that none of the errors changed the conclusions of the safety evaluation screenings. This condition report was categorized as a “B” level corrective action report by licensee personnel and requires an apparent or root cause analysis to be done. Licensee personnel have not completed the cause analysis nor established corrective actions. This finding was of very low safety significance and was assessed as a Severity Level IV NCV.

.2 Controlled Ventilation System Post-LOCA Analysis

10 CFR Part 50, Appendix B, Criterion III, “Design Control,” states, in part, that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to this, Entergy identified that several design base analyses failed to consider the post-LOCA filter loading on the CVAS filters. This finding is of very low safety significance because it did not result in loss of equipment safety function. This was identified in Entergy’s corrective action program as Condition Reports CR-WF3-2004-2461, CR-WF3-2004-3560, and CR-WF3-2004-3573.

.3 Adequacy of ASME Section XI Testing for Safety Injection System Valves

10 CFR Part 50, Appendix B, Criterion III, “Design Control,” states, in part, to establish measures to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to that, Entergy identified that the design function of safety injection Valves SI-602A and SI-602B was not translated into testing procedures and instructions, resulting in the failure to perform required ASME Section XI In-Service Testing. This finding is of very low safety significance because it did not result in loss of equipment function. This deficiency was identified in Entergy’s corrective action program as Condition Report CR-WF3-2004-02847.

.4 (Closed) URI 05000382/2004004-06: Review Safety Significance of Safety Injection Valve SI-602B Leakage

10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” requires that testing be performed to demonstrate that components will perform satisfactorily in service. On November 6, 2003, following adjustments made to SI Valve SI-602B, Entergy failed to perform a leak test to identify if the adjustments affected the leak tightness of the valve seat. Subsequently, on September 9, 2004, during leak testing, it was identified that Valve SI-602B exhibited excessive leakage. The excessive leakage created a condition that could potentially result in premature closure of the RWSP downstream check valve, resulting in loss of suction to the Train B emergency core cooling system (ECCS) and containment spray pumps. A Phase 2 analysis was performed assuming loss of these components for all accident scenarios involving containment pressurization. The result of this analysis resulted in a risk assessment that would be of greater than very low safety significance. As a result, a Phase 3 analysis was performed by a senior reactor analyst. After additional analysis was performed, it was determined that Train B ECCS and containment spray pump failure was probable for medium and large break LOCAs. Based on containment pressure profiles for the small break LOCA, it was concluded that sufficient pressure to prematurely close the RWSP downstream check valve was not available; therefore, the ECCS and containment spray safety functions were not adversely affected. Given the low probability of a medium or large break LOCA scenarios, this finding was determined to be of very low safety significance.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. S. Anders, Superintendent, Plant Security
J. Brawley, ALARA Coordinator, Radiation Protection
N. T. Brumfield, Manager, Quality Assurance
K. Cook, Manager, Systems Engineering
L. Dauzat, Supervisor, Radiation Protection
R. A. Dodds, Manager, Plant Licensing Staff
C. Fugate, Assistant Manager, Operations (Shift)
A. Harris, Manager, Engineering Projects
J. Holman, Manager, Nuclear Engineering
B. Houston, Manager, Radiation Protection
P. Kelly, Supervisor, Radiation Protection
B. Lanka, Supervisor, Design Engineering
J. Laque, Manager, Maintenance
J. J. Lewis, Manager, Emergency Preparedness
R. Madjerich, Manager, Operations
M. Mason, Technical Specialist IV, Licensing
T. Mitchell, Director, Engineering
R. Murillo, Senior Staff Engineer, Licensing
R. Osborne, Manager, Programs and Components
K. Peters, Director, Nuclear Safety Assurance
R. Peters, Manager, Planning and Scheduling
B. Pilutti, Supervisor, Radiation Protection
R. Porter, Technical Assistant
J. Rachal, Supervisor, Design Engineering
J. Reese, Manager, Design Engineering
G. Scott, Licensing Engineer
R. Sebring, Senior Health Physics Technician, Radiation Protection
C. Stafford, Manager, Corporate Assessments
C. Tacazar, Supervisor, System Engineering
J. Venable, Vice President, Operations
K. T. Walsh, General Manager, Plant Operations
R. Williams, Licensing Engineer

NRC

M. Hay, Senior Resident, Waterford
G. Larkin, Resident, Waterford
L. Ricketson, Senior Health Physicist (via telephone)

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000382/2004005-01	NCV	Inadequate Test Controls to Identify Leakage of Potentially Highly Radioactive Fluids Outside Containment (Section 1R22)
05000382/2004005-02	NCV	Technical Specification Violation for Failure to Follow Radiation Work Permit Requirements (Section 2OS1)
05000382/2004005-03	NCV	Failure to Prevent a Reoccurrence of an Over-Pressure Condition in Main Feed Water Isolation Valve Hydraulic Operating Systems (Section 4OA2)
05000382/2004005-04	NCV	Failure to Isolate Neutral Wire during Transfer to the Alternative Shutdown Panel per 10 CFR Part 50, Appendix R, Section III.L.3 (Section 4OA5)

Closed

05000382/2004005-01	NCV	Inadequate Test Controls to Identify Leakage of Potentially Highly Radioactive Fluids Outside Containment (Section 1R22)
05000382/2004005-02	NCV	Technical Specification Violation for Failure to Follow Radiation Work Permit Requirements (Section 2OS1)
05000382/2004005-03	NCV	Failure to Prevent a Reoccurrence of an Over-Pressure Condition in Main Feed Water Isolation Valve Hydraulic Operating Systems (Section 4OA2)
05000382/2004005-04	NCV	Failure to Isolate Neutral Wire during Transfer to the Alternative Shutdown Panel per 10 CFR Part 50, Appendix R, Section III.L.3 (Section 4OA5)
05000382/2004004-04	URI	Instrument Uncertainties for Auxiliary Component Cooling Water System (Section 4OA5)
05000382/2004004-06	URI	Review Safety Significance of Safety Injection Valve SI-602B Leakage (Section 4OA5)
05000382/2000007-04	URI	Alternate shutdown panel may not be electrically isolated from a control room fire due to multiple spurious actuations (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R02: <u>Evaluation of Changes, Tests, or Experiments</u>

Condition Reports

CR-WF3-2002-00175, CR-WF3-2002-00322, CR-WF3-2002-00601, CR-WF3-2002-00811, CR-WF3-2002-00818, CR-WF3-2002-00858, CR-WF3-2002-00899, CR-WF3-2002-02038, CR-WF3-2002-02042, CR-WF3-2003-00299, CR-WF3-2004-01030, CR-WF3-2004-01335, CR-WF3-2004-01606, CR-WF3-2004-01865, CR-WF3-2004-01869, CR-WF3-2004-01892 CR-WF3-2004-02859, CR-WF3-2004-02861, CR-WF3-2004-03216, CR-WF3-2004-03217, CR-WF3-2004-03293, CR-WF3-2004-03337, and CR-WF3-2004-03339

Miscellaneous

Containment Isolation and Leakage Rate Testing, Revision 7
Inservice Test Bases Document, Revision 3
WA 01153606, "SI-125A&B and SI-412A&B Pressure Equalization Line Additions"

Procedures

LI-101, "10 CFR 50.59 Review Program," Revision 4
DG-LI-101, "10 CFR 50.59 Review Program Guidelines," Revision 6
ENS-DC-115, "Engineering Requests Response Development," Revision 5
OP-100-009, "Control of Valves and Breakers," Revision16, Change 3

Safety Evaluation Applicability Reviews

ER-W3-1998-0359-00-00	ER-W3-2004-0008-000	ER-W3-2004-0373-000
ER-W3-2001-0312-000	ER-W3-2004-0128-000	ER-W3-2004-0373-003
ER-W3-2002-0198-000	ER-W3-2004-0271-000	ER-W3-2004-0382-000
ER-W3-2003-0142-000	ER-W3-2004-0295-000	ER-W3-2004-0396-000
ER-W3-2004-0418-000	ER-W3-2004-0419-000	ER-W3-2004-0310-001
ER-W3-2004-0624-000	ER-W3-2004-0298-000	ER-W3-2004-0325-001

Safety Evaluations

03-001	03-006	03-008	03-013-1
03-005	03-007	03-011	

Safety Evaluation Screenings

ER-W3-97-0451-00-00	ER-W3-2001-1126-000	ER-W3-2004-0396-001
ER-W3-97-0456-00-00	ER-W3-2004-0115-001	ER-W3-2004-0439-001
ER-W3-98-0137-00-00	ER-W3-2004-0116-001	
ER-W3-2000-0599-000	ER-W3-2004-0217-001	
ER-W3-2000-0599-000		

Section 1R04: Partial System Walkdown

Procedures
OP-009-008, “Safety Injection System,” Revision 16

Condition Reports

CR-WF3-2004-3853	CR-WF3-2004-3307	CR-WF3-2004-1558
CR-WF3-2004-3596	CR-WF3-2004-0011	CR-WF3-2003-4010
	CR-WF3-2004-2176	

Section 1R05: Fire Protection

Procedures:

NUMBER	TITLE	REVISION
Maintenance Procedure 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

Section 1R06: Flood Protection Measures

Procedures

NUMBER	TITLE	REVISION
OP-903-521	Severe Weather and Flooding	3
OP-002-007	Freeze Protection and Temperature Maintenance	11

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
Calculation MN(Q)-3-5	Flooding Analysis Outside Containment	3

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
FSAR Section 3.4	Water Level Flood Design	
FSAR Section 2.4.2	Floods	

Procedures

NUMBER	TITLE	REVISION
OI-037-000	Operations' Risk Assessment Guidelines	0

Section 1R15: Operability Evaluations

DOCUMENT	TITLE/SUBJECT	REVISION
ESC04-017	Control Room Dose Due to Filter Loading Using Alternate Source Term (AST) Methodology	0
EC-S96-011	LOCA Offsite and Control Room Radiological Dose Consequences	1
ER-W3-2004-0566-000	ESF Leakage Rate Margin Recovery	0
EC-S96-002	Post-LOCA Dose Due to ESF System Leakage	1
EC-S04-018	RAB +46 Filter Shine Dose Effects - RAB HVAC Room EQ Doses	0
ER-W3-00-0686	Raise the Bearing and Discharge Trip Setpoints on All Three Essential Chillers to Preclude spurious Trips and Failures to Start During SIAS and LOP Modes of Operation	1

DOCUMENT	TITLE/SUBJECT	REVISION
RAC26	Loadings on Control Room Charcoal Filters	0
3-E-1	CVAS Areas: Exhaust Requirements	0

Section 1R17: <u>Permanent Plant Modifications</u>

<u>Design Changes</u>

NUMBER	TITLE	REVISION
ER-W3-1999-1057-000	Reactor Trip Breaker Sure Trip Overcurrent Device	0
ER-W3-2000-0686-000	Raise the Bearing and Discharge Trip Setpoints on All Three Essential Chillers to Preclude Spurious Trips and Failures to Start During SIAS and LOP Modes of Operation	1
ER-W3-2000-0991-00-00	Essential Chiller Reliability Enhancements	0
ER-W3-2001-0305-000	Reactor Head Assembly Improvements - CEDM Cooling System Modification	0
ER-W3-2001-0305-001	Installation of Permanent Cavity Seal Ring	0
ER-W3-2001-0379-000	EGA-140 A&B and EGA-141 A&B Relief Valve Replacement	0
ER-W3-2001-0399-000	Bypassing of the EDG-A MIN-MAX Excitation Limiter	0

Design Changes

NUMBER	TITLE	REVISION
ER-W3-2001-0404-000	Modification to Disable the Emergency Diesel Generator Voltage Regulator Min-Max Excitation Limiter	0
ER-W3-2002-0300-000	Change of Wiring in Essential Chillers Due to Relay Race Conditions	0
ER-W3-2002-0323-000	Core Protection Calculator Trip Upon a Loss of Power (Watchdog Timer Modification)	0
ER-W3-2003-0161-000	CHW-ITE-5022C (Conduit 31072D-SAB) - Resolve Appendix R Issue	0
ER-W3-2003-0457-000	EDG Air Dryer Interlocked Operation	0

Section 1R19: Postmaintenance Testing

Condition Reports

CR-WF3-2004-04045	CR-WF3-2004-04172	CR-WF3-2004-04093
CR-WF3-2004-03788	CR-WF3-2004-04067	
CR-WF3-2002-00086	CR-WF3-2004-02197	

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
47000093	Charging Pumps	
WO 00057486	FWIV #1 Accumulator A Pressure Restoration	0

Section 1R22: Surveillance Testing

Procedures

NUMBER	TITLE	REVISION
OP-903-110	RAB Fluid System Leak Test	5
OP-903-024	Reactor Coolant System Water Inventory Balance	13
OI-040-000	Reactor Coolant System Leakage Monitoring	0
W3-DBD-026	Containment Isolation and Leak Testing	0
CEP-IST-1	IST Bases Document	3

Condition Reports

CR-WF3-2004-03601	CR-WF3-2004-02847	CR-WF3-2004-03457
CR-WF3-2004-00252	CR-WF3-2004-03977	CR-WF3-2004-04040
CR-WF3-2004-03888	CR-WF3-2004-03454	CR-WF3-1998-01008

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
IE Compliance Bulletin 86-03	Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line	0

Section 1R23: Temporary Plant Modifications

Procedures

NUMBER	TITLE	REVISION
UNT-005-004	Temporary Alteration Control	16
ESC04-017	Control Room Dose Due to Filter Loadint Using Alternate Source Term (AST) Methodology	0
EC-S96-011	LOCA Offsite and Control Room Radiological Dose Consequences	1

Procedures

NUMBER	TITLE	REVISION
ER-W3-2004-0566-000	ESF Leakage Rate Margin Recovery	0
EC-S96-002	Post-LOCA Dose Due to ESF System Leakage	1
EC-S04-018	RAB +46 Filter Shine Dose Effects - RAB HVAC Room EQ Doses	0
ER-W3-00-0686	Raise the Bearing and Discharge Trip Setpoints on All Three Essential Chillers to Preclude spurious Trips and Failures to Start During SIAS and LOP Modes of Operation	1
RAC26	Loadings on Control Room Charcoal Filters	0
3-E-1	CVAS Areas: Exhaust Requirements	0

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
TSR 2004-0047	Protect Chiller's A and B from Radiation Degradation	0

Section 1EP2: Alert Notification System Testing

Procedures
Procedure EPP-422, "Siren and Helicopter Warning System Maintenance," Revision 2
Procedure EPP-424, "Siren Testing and Siren System Administrative Controls," Revision 7
Desk Guide 16, "Siren System Administrative Data," Revision 9

Section 1EP3: Emergency Response Augmentation Testing

Procedures

Procedure EP-002-100, "Technical Support Center Activation, Operation, and Deactivation," Revision 31

Procedure EP-002-101, "Operational Support Center Activation, Operation, and Deactivation," Revision 27

Procedure EP-002-102, “Emergency Operations Facility Activation, Operation, and Deactivation,” Revision 27

Notification Pager Drills

January 7, 2003	August 18, 2003	March 27, 2004
February 24, 2003	September 10, 2003	April 27, 2004
March 28, 2003	October 17, 2003	May 27, 2004
April 27, 2003	November 23, 2003	June 14, 2004
May 22, 2003	December 15, 2003	July 21, 2004
June 30, 2003	January 22, 2004	September 29, 2004
July 16, 2003	February 15, 2004	

Section 1EP5: Correction of Emergency Preparedness Weaknesses and Deficiencies

Quality Assurance Surveillance Report QS-2003-W3-011

Quality Assurance Surveillance Report QS-2004-W3-005, Emergency Plan Drill, March 11, 2004

Performance Indicator Self Assessment, May 2003

Performance Indicator Self-Assessment, August 2004

South Texas Project Emergency News Center Benchmark Analysis

Drill Reports: 2002-09, 2002-05, 2003-01, 2003-04, 2003-05, W3D3-04-0001, 2004-01, 2004-04

Condition Reports

CR-WF3-2002-1815 and 1875
CR-WF3-2003-0772, 1024, 1104, 1838, 1841, 2091, 2396, 3760, 3885, and 3907
CR-WF3-2004-0556, 0721, 0965, 1029, 1898, 2064, 2113, 2244, 2857, and 3005

Quality Assurance, Oversight Observation Checklists

January 12, 2004	March 29, 2004	September 29, 2004
February 19, 2004	August 5, 2004	September 30, 2004
March 10, 2004	August 12, 2004	October 2, 2004
March 11, 2004	August 23, 2004	October 5, 2004
March 12, 2004		

Section 2OS1: Access Control To Radiologically Significant Areas (IP71121.01)

Radiation Work Permits

2003-1509	Steam Generator 1 and 2 Primary Side Work
2003-1510	Install and Remove Steam Generator Nozzle Dams
2003-1613	Replacement of Pressurizer Heaters

2003-1713	Work involving Non-Destructive Examination under Reactor Head
2003-1705	Reactor Re-Assembly

Procedures

UNT-001-016	Radiation Protection Program, Revision 1
RP-102	Radiological Controls, Revision 3
RP-108	Radiation Protection Postings, Revision 2
HP-001-123	Plant Conditions and Radiological Concerns, Revision 2
HP-001-107	High Radiation Area Access Control, Revision 16
HP-002-221	Fuel Transfer Shield Survey, Revision 4
HP-002-222	Steam Generator Radiological Controls, Revision 6
HP-002-224	Spent Resin Operations, Revision 4
HP-002-215	Airborne Survey Techniques, Revision 4
HP-001-243	Diving Operations in Contaminated Waters Near Highly Radioactive Components, Revision 6
EN-LI-114	Performance Indicator Process, Revision 0

Condition Reports

2003-3164, 2003-3238, 2003-3252, 2003-3314, 2003-3439, 2003-3443, 2003-3501,2003-3550, 2003-3551, 2003-3583, 2003-3615, 2003-4001, 2004-0845, 2004-1815, 2004-3114, and 2004-3316

Self-Assessments/Audits

LO-WLO-2004-0088	RP Corporate Assisted Self-Assessment
Snapshot Assessment	Compensatory Actions in High Noise Areas and Implementation of SOER-01-1 Recommendation 6b

Miscellaneous

2003 Annual Radioactive Effluent Report

Section 2OS2: ALARA Planning and Controls

Corrective Action Documents
CR-WF3-2003-03793, CR-WF3-2004-00093, CR-WF3-2004-00102, CR-WF3-2004-00347, CR-WF3-2004-00401, CR-WF3-2004-00738, CR-WF3-2004-00978, CR-WF3-2004-01082, CR-WF3-2004-02100, CR-WF3-2004-02290, CR-WF3-2004-02397, CR-WF3-2004-02741, and CR-WF3-2004-03382

Audits and Self-Assessments
ALARA Planning and Controls, WLO-2004-0093 CA 1, July 26-29, 2004

Annual Radiation Protection Report for Waterford 3 SES Plant Year 2003

Monthly Radiation Protection Report, October 2004

Snapshot Assessment, Compensatory Actions in High Noise Areas and Implementation of SOER-01-1 Recommendation 6b, May 24, 2004

TLD Processing Annual Assessment, LO-WLO-2004-00081 CA-01, June 28-30, 2004

Shielding Requests
2001-01
2001-03

Radiation Work Permits
2003-1702
2004-1011

Procedures
HP-001-114, Installation of Temporary Shielding
RP-102, Radiological Control
RP-105, Radiation Work Permits
RP-109, Hot Spot Program
RP-110, ALARA Program
RP-205, Prenatal Monitoring
UNT-001-016, Radiation Protection

ALARA Committee Minutes

December 2003 through October 2004

Section 4OA1: Performance Indicators

Procedures

Procedure EP-001-001, “Recognition and Classification of Emergency Conditions,” Revision 19
Procedure EP-002-010, “Notifications and Communications,” Revision 29
Procedure EP-002-052, “Protective Action Guidelines,” Revision 18

Miscellaneous Documents

Desk Guide 02, “Control Room Check,” Revision 2
Desk Guide 03, “TSC Check,” Revision 7
Desk Guide 04, “EOF/Backup EOF Check,” Revision 5
Desk Guide 06, “Operational Support Center,” Revision 6
Desk Guide 13, “Public Information Material Updates,” Revision 3
Desk Guide 19, “Offsite Facilities Check,” Revision 0

Section 4OA2: Identification and Resolution of Problems

Condition Reports

CR-WF3-2004-04172	CR-WF3-2004-04124	CR-WF3-2004-03293
CR-WF3-2004-04067	CR-WF3-2004-01606	CR-WF3-2004-03337
CR-WF3-2004-02197	CR-WF3-2004-03216	CR-WF3-2004-03339
CR-WF3-2004-04093	CR-WF3-2004-03217	

Procedure EP-001-001, “Recognition and Classification of Emergency Conditions,” Revision 19
Procedure EP-002-010, “Notifications and Communications,” Revision 29
Procedure EP-002-052, “Protective Action Guidelines,” Revision 18

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Desk Guide 04, “EOF/Backup EOF Check,” Revision 5
Desk Guide 06, “Operational Support Center,” Revision 6
Desk Guide 13, “Public Information Material Updates,” Revision 3
Desk Guide 19, “Offsite Facilities Check,” Revision 0

LIST OF ACRONYMS AND ABBREVIATIONS

ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
ACCW	auxiliary component cooling water
CCW	component cooling water
CFR	*Code of Federal Regulations*
CS	containment spray
CVAS	controlled ventilation area system
ECCS	emergency core cooling system

Entergy	Entergy Operations, Inc.
LOCA	loss of coolant accident
LPSI	low pressure safety injection
MFIV	main feedwater isolation valve
NCV	noncited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
RWSP	reactor water storage pool
SDP	significance determination process
SI	safety injection
UFSAR	Updated Final Safety Analysis Report
URI	unresolved item
Waterford 3	Waterford Steam Electric Station