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

February 2, 2004

EA 03-230

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

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000382/2003007

Dear Mr. Venable:

On December 31, 2003, 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 5, 2003, 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.

The report discusses a finding that appears to have Greater than Green safety significance. As described in Section 4OA3.1 of this report, the issue involved the failure to establish appropriate instructions and accomplish those instructions for installation of a fuel line for Train A emergency diesel generator in May of 2003. This failure resulted in uneven and excessive scoring of the tubing that ultimately led to a complete 360 degree failure of the fuel supply line on September 29, 2003, during a monthly surveillance test, which rendered the Train A emergency diesel generator inoperable. This finding was assessed based on the best available information, including influential assumptions, using the applicable Significance Determination Process and was preliminarily determined to be a Greater than Green Finding. The final resolution of this finding will convey the increment in the importance to safety by assigning the corresponding color i.e, White (a finding with some increased importance to safety, which may require additional NRC inspection), Yellow (a finding with substantial importance to safety that will result in additional NRC inspection and potentially other NRC action) or Red (a finding of high importance to safety that will result in increased NRC inspection and other NRC action). Because the preliminary safety significance is greater than Green, the NRC requests that additional information be provided regarding the nonrecovery probability for the Train A emergency diesel generator and any other considerations you have identified as impacting the safety significance determination.

This finding does not present an immediate safety concern based on your immediate and long term corrective actions. These actions included a complete re-design and installation of the fuel line that had prematurely failed.

This finding is also an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The current enforcement policy is included on the NRC's website at http://www.nrc.gov/what-we-do/regulatory/enforcement.html.

Before the NRC makes a final decision on this matter, we are providing you an opportunity (1) to present to the NRC your perspectives on the facts and assumptions, used by the NRC to arrive at the finding and its significance, at a Regulatory Conference or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of the receipt of this letter.

Please contact Mr. William Jones at (817) 860-8147 within 10 days of the date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision and you will be advised by separate correspondence of the results of our deliberations on this matter.

Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for the inspection finding at this time. In addition, please be advised that the characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

In addition, the enclosed report documents four NRC-identified findings of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these four findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the 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 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure(s), 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/*

Arthur T. Howell III, Director
Division of Reactor Projects

Docket: 50-382
License: NPF-38

Enclosure:
NRC Inspection Report
050000382/2003007

w/attachment: Supplemental Information

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| RIV:RI:DRP/E | SRI:DRP/E | C:DRS/PSB | C:DRS/OB |
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| GFLarkin | MCHay | TWPruett | ATGody |
| ***T - WBJones*** | ***T - WBJones*** | ***E - WBJones*** | ***/RA/*** |
| 1/28/04 | 1/28/04 | 1/28/04 | 1/27/04 |
| **C:DRS/EB** | **SRA** | **C:DRP/E** | **D:DRP** |
| CSMarschall | MFRunyan | WBJones | ATHowell |
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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

| | |
|---|---|
| Docket: | 50-382 |
| License: | NPF-38 |
| Report: | 05000382/2003007 |
| Licensee: | Entergy Operations, Inc. |
| Facility: | Waterford Steam Electric Station, Unit 3 |
| Location: | Hwy. 18<br>Killona, Louisiana |
| Dates: | September 21 through December 31, 2003 |
| Inspectors: | M. C. Hay, Senior Resident Inspector<br>G. F. Larkin, Resident Inspector<br>M. P. Shannon, Senior Health Physicist<br>P. C. Gage, Senior Operations Engineer<br>T. O. McKernon, Senior Operations Engineer<br>J. F. Drake, Operations Engineer<br>V.G. Gaddy, Senior Project Engineer<br>D. R. Carter, Health Physicist<br>W. C. Sifre, Reactor Inspector, Engineering Branch<br>T. McConnell, Reactor Inspector, Engineering Branch |
| Accompanying Personnel: | V. X. Thomas, Intern, NRC Headquarters |
| Approved By: | A. T. Howell III, Director, Division of Reactor Projects |
| ATTACHMENT: | Supplemental Information |

Enclosure

## CONTENTS

## SUMMARY OF FINDINGS

IR05000382/2003-007; 09/21/2003-12/27/2003; Waterford Steam Electric Station, Unit 3; Postmaintenance Testing, Access Control to Radiological Significant Areas, Identification and Resolution of Problems, and Event Followup.

The report covered a 15-week period of inspection by resident inspectors, a senior health physicist, a health physicist, two senior operations engineers, an operations engineer, a senior project engineer, and a reactor engineer**.** The inspection identified one potential greater than green finding and four green findings. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 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

- TBD. A self-revealing apparent violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for the failure to establish appropriate instructions and accomplish those instructions for installation of a fuel line for Train A emergency diesel generator in May 2003. This failure resulted in uneven and excessive scoring of the tubing that ultimately led to a complete 360 degree failure of the fuel supply line on September 29, 2003, during a monthly surveillance test.

  This finding is unresolved pending completion of a significance determination. The finding was greater than minor because it directly impacted the availability and reliability of an emergency diesel generator which is used to mitigate the loss of AC power to the respective safety related bus. The finding was determined to have a potential safety significance greater than very low significance because the failure resulted in an actual loss of the safety function of the Train A emergency diesel generator for an extended period of time (Section 4OA3).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to establish adequate corrective actions to prevent recurrence of voiding conditions affecting the operability of the low pressure safety injection system following shutdown cooling operations.

  This finding is greater than minor because it affected the mitigating system objective to ensure the reliability and availability of the low pressure safety injection system to respond to an initiating event. The problem if left uncorrected would become a more significant safety concern. The significance of this finding was determined to be of very low safety significance because low pressure safety injection Train B was inoperable for less than the Technical Specification allowed outage time and Train A was determined to be degraded but operable in accordance with Generic Letter 91-18 guidance (Section 4OA2).

Cornerstone: Barrier Integrity

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Section XI, "Test Control," for the failure to establish adequate test controls for leak testing main steam isolation Valves 1 and 2. This performance deficiency contributed to both valves being declared inoperable due to system leaks creating a low pressure condition in the valve actuating systems.

  This finding is more than minor because it affected the Barrier Integrity Cornerstone objective of providing reasonable assurance of the functionality of containment. 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, it did not result in an actual open pathway affecting the physical integrity of reactor containment, and the main steam isolation valves were inoperable for less time than the allowed Technical Specification outage time (Section 1R19).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to implement effective corrective actions resulting in recurrences of pressure boundary leakage due to primary water stress corrosion cracking of Alloy 600 reactor coolant system nozzles.

  This finding was greater than minor because it affected the reactor safety barrier integrity cornerstone objective for providing reasonable assurance that the physical design barriers protect the public from radionuclide releases caused by accidents or events. Using NRC Manual Chapter 0609 Significance determination process Phase 1 Screening Worksheet this performance deficiency affected the reactor coolant system barrier function requiring a Phase 2 analysis. The results of the Phase 2 and 3 analysis determined that this finding was of very low safety significance based on the cracks being axial in nature (does not contribute substantially to a loss of coolant accident) and the leaks resulted in a build up of only minor boric acid residue indicative of only trace amounts of through wall leakage. The leak rates identified were well within the capacity of a single charging pump (4OA2).

Cornerstone: Occupational Radiation Safety

- Green. The inspector identified a noncited violation of Technical Specification 6.12.1 because Entergy failed to barricade a high radiation area. Specifically, on October 27, 2003, the inspector observed that the high radiation area rope barricading the regenitive heat exchanger room was stretched across the entrance way at a height of approximately 79 inches, which would not obstruct the entry of station workers. General area radiation levels within the room were as high as 420 millirem per hour. The finding is in Entergy's corrective action program as Condition Report CR-WF3-2003-03164.

The finding is greater than minor because it affected the Occupational Radiation Safety cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation and the finding is associated with the cornerstone attribute (Program & Process). The finding involved an individual's potential for unplanned or unintended dose. When processed through the Occupational Radiation Safety Significance Determination Process the finding was determined to be of very low safety significance because the finding was not associated with ALARA planning or work controls, there was no overexposure or a substantial potential for overexposure, and the ability to assess dose was not compromised (Section 2SO1).

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 began the period on September 21, 2003, at 97 percent power and coasted down to 75 percent power on October 20, 2003. The plant was shutdown for a scheduled refueling outage on October 20, 2003. On November 20, 2003, operators commenced a reactor startup to perform low power physics testing. The main turbine generator was placed online on November 22, 2003, and the refueling outage ended on November 24, 2003. Power was increased and reached approximately 100 percent on November 25, 2003. Power remained at that level until December 19, 2003, when power was reduced to 95 percent power for moderator temperature coefficient testing. Following testing, power was increased to 100 percent.

## 1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

### 1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors completed one adverse weather protection inspection during this inspection period. On December 19, 2003, the inspectors completed a walkdown of components and systems susceptible to freezing using Procedure OP-002-007, "Freeze Protection and Temperature Maintenance," Revision 10, to verify that the onset of cold weather would not affect the mitigating systems. This inspection included a review of condition reports associated with heat tracing and other cold weather protection measures to determine their impact on the systems. Additionally, the inspectors discussed adverse weather preparations with various Entergy Operations, Inc. (Entergy) personnel.

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 two partial system equipment alignment inspections during this inspection period:

- On November 3, 2003, the inspectors walked down the accessible portions of the spent fuel pool cooling system, Train A. The walkdown was completed following a full core offload to verify that cooling water flow to the spent fuel pool was adequate to maintain adequate cooling for the spent fuel. The inspectors performed the walkdown using Procedure OP-002-006, "Fuel Pool Cooling and Purification," Revision 15. The inspectors had reviewed the ability of the spent fuel pool cooling system to remove the decay heat of the spent fuel during refueling outages involving a full core offload during the previous inspection period (NRC Inspection Report 05000382/2003006) and as documented in Section 1R07 to this report.

- On December 18, 2003, the inspectors performed a partial equipment alignment inspection of the reactor auxiliary building cable vault area and switchgear area ventilation system Train B while the switchgear area ventilation system Train A was inoperable. A review of selected maintenance work orders and corrective action documents was performed to assess the material condition and performance of the system. System configuration was assessed using Operating Procedure OP-003-026, "Cable Vault and Switchgear HVAC," Revision 7. 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.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdowns

a. Inspection Scope

The inspectors performed a complete alignment inspection of the safety related nitrogen system. A walkdown of the mechanical and electrical components in the system was performed to verify that the system was configured and operated in accordance with Operating Procedure OP-003-019, "Nitrogen System," Revision 12. The inspectors reviewed the nitrogen system design requirements in the Updated Final Safety Analysis Report to verify the system's ability to provide back up source of compressed gas for various safety-related valves was adequate. The inspectors reviewed Engineering Request ER-W3-97-0547-000 and select condition reports written on the nitrogen system since October 1, 2000, to verify that degraded conditions were identified at the appropriate threshold and that corrective actions were 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 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 were inspected:

- Fire Zone RAB 2, 15, 16, 17 and 18 on October 1, 2003
- Fire Zone RAB 1A, 1B, 5, 6 and 7 on October 6, 2003
- Fire Zone RAB 35, 36, 37, 38 and 39 on November 20, 2003
- Fire Zone RAB 1A, 8, 11, 12 and 13 on November 28, 2003
- Fire Zone RAB 1A, 5, 6, 7 and 8 on December 18, 2003
- Fire Zone RAB 35, 36, 37, 38, and 39 on December 30, 2003

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

As discussed in NRC Inspection Report 05000382/200306, Section 1R07, the inspectors previously reviewed documentation, analysis, and design basis documentation relative to the ability of the spent fuel pool cooling system to remove decay heat of the spent fuel during refueling outages involving a full core offload. During this inspection period (as discussed in Section 1R04 of this report) on November 3, 2003, the inspectors walked down the accessible portions of the spent fuel pool cooling system Train A. The walkdown was completed following a full core offload to verify that cooling flow to the spent fuel pool was adequate to maintain spent fuel pool temperature.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Nondestructive Examination Activities

The inspectors observed the ultrasonic system calibration and observed ultrasonic and magnetic particle examinations. The inspectors observed 13 examinations, which are listed below.

| System | Component/Weld Identification | Examination Method |
|---|---|---|
| RCS cold leg | 07-013 | Ultrasonic |
| RCS cold leg | 07-013 | Magnetic Particle |
| RCS cold leg | 07-016 | Ultrasonic |
| RCS cold leg | 07-016 | Magnetic Particle |
| RCS Hot leg | 06-010 | Magnetic Particle |
| RCS Hot leg | 06-010 | Ultrasonic |
| RCS Hot leg | 06-011 | Magnetic Particle |
| RCS Hot leg | 06-011 | Ultrasonic |
| Main Feed Header B | 46-011 | Ultrasonic |
| Main Feed Header B | 46-012 | Ultrasonic |
| Main Feed Header B | 46-020 | Ultrasonic |
| #2 Steam Generator Nozzle | 04-030 | Magnetic Particle |
| Containment Penetration | 55-050 | Ultrasonic |

During the review of these examinations, the inspectors verified that the correct nondestructive examination (NDE) procedure was used, examinations and conditions were as specified in the procedure, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors also reviewed the documentation to determine if the indications revealed by the examinations were compared against the American Society of Mechanical Engineers (ASME) Code specified acceptance standards, and that the indications were appropriately dispositioned. The nondestructive examination certifications of those personnel observed performing examinations or identified during review of completed examination packages were reviewed by the inspectors.

The inspectors also observed the ultrasonic and eddy current examination of two leaking pressurizer heater sleeves. In each case the flaws were clearly identified as short axial flaws near the sleeve/pressurizer interface. The two pressurizer heater sleeves were repaired using a MNSA-2 clamp.

b. Findings

No findings of significance were identified.

.2 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspection procedure specified, with respect to in situ pressure testing, performance of an assessment of in situ screening criteria to assure consistency between assumed NDE flaw sizing accuracy and data from the Electric Power Research Institute (EPRI) examination technique specification sheets. It further specified assessment of appropriateness of tubes selected for in situ pressure testing, observation of in situ pressure testing, and review of in situ pressure test results.

The inspectors selected and reviewed the acquisition technique sheets and their qualifying EPRI examination technique specification sheets to verify that the essential variables regarding flaw sizing accuracy had been identified and qualified through demonstration.

The inspection procedure specified comparing the estimated size and number of tube flaws detected during the current outage against the previous outage operational assessment predictions to assess Entergy's prediction capability. The inspectors reviewed Report ER-W3-2003-0534-000, "Steam Degradation Assessment and Repair Criteria for RF12." The purposes of the report were: (1) to provide a comprehensive review and overall plan for detection and assessment of degradation to be addressed during Refueling Outage RF12; and, (2) to provide predictions as to the type and extent of degradation expected to be found.

The inspection procedure specified confirmation be made that the steam generator tube eddy current testing (ECT) scope and expansion criteria meet technical specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors review determined that the steam generator tube ECT scope and expansion criteria were being met.

The inspection procedure also specified that, if Entergy identified new degradation mechanisms, then verify that Entergy had fully enveloped the problem in an analysis and had taken appropriate corrective actions before plant startup. At the time of this inspection, no new degradation mechanisms had been identified.

The inspection procedure also required confirmation that all areas of potential degradation were being inspected, especially areas which were known to represent potential ECT challenges (e.g., top-of-tubesheet, tube support plates, and U-bends). The inspectors confirmed that all known areas of potential degradation, including ECT challenged areas, were included in the scope of inspection and were being inspected.

The inspection procedure further required that repair processes being used were approved in the technical specifications for use at the site. At the time of this inspection, Entergy had not performed or used the designated Technical Specification approved repair processes, thus there was no opportunity to observe implementation of any potential repairs (e.g., plugging operations) or in-situ pressure testing.

The inspection procedure also required confirmation that the Technical Specification plugging limit was being adhered to, and determination whether depth sizing repair criteria were being applied for indications other than wear or axial primary water stress corrosion cracking in dented tube support plate intersections. The inspectors confirmed that Entergy was adhering to these specifications. The inspectors also determined that Entergy, in response to Information Notice 2002-21, did account for crack-like indications in dented tube support plate intersections by including these parameters in their ECT computer programming, and the acquisition and analysis technique sheets. Further, the ECT data analysts had been presented with specialized training associated with this type of indication.

The inspection procedure stated that if steam generator leakage greater that three gallons per day was identified during operations or during postshutdown visual inspections of the tubesheet face, then assess whether Entergy had identified a reasonable cause and corrective actions for the leakage based on inspection results. The inspectors did not conduct any assessment because this condition did not exist.

The inspection procedure required confirmation that the ECT probes and equipment were qualified for the expected types of tube degradation and assessment of the site specific qualification of one or more techniques. The inspectors observed portions of all ECT performed. During these examinations, the inspectors verified that: (1) the probes appropriate for identifying the expected types of indications were being used; (2) probe position location verification was performed; (3) calibration requirements were adhered to; and, (4) probe travel speed was in accordance with procedural requirements. The assessment of site specific qualifications of the techniques being used, including a listing of the specific techniques and qualifications reviewed, is addressed and identified in the table above.

Finally, the inspection procedure specified the review of one to five samples of ECT data if questions arose regarding the adequacy of ECT data analyses. The inspectors did not identify any results where ECT data analyses adequacy was questionable.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed inservice inspection related condition reports issued during the current and past refueling outage, and verified that Entergy identified, evaluated, corrected, and trended problems. In this effort, the inspectors evaluated the effectiveness of Entergy's corrective action process, including the adequacy of the technical resolutions.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

.1 Biennial Inspection

a. Inspection Scope

The inspectors: (1) evaluated examination security measures and procedures for compliance with 10 CFR 55.49; (2) evaluated Entergy's sample plan of the written examinations for compliance with 10 CFR 55.59 and NUREG-1021, as referenced in the facility requalification program procedures; and (3) evaluated maintenance of license conditions for compliance with 10 CFR 55.53 by review of facility records (medical and administrative), procedures, and tracking systems for licensed operator training, qualification, and watchstanding. In addition, the inspectors reviewed remedial training for examination failures for compliance with facility procedures and responsiveness to address areas failed.

Furthermore, the inspectors: (1) interviewed 12 personnel, including operators, instructors/evaluators, and training supervisors, regarding the policies and practices for administering requalification examinations; (2) observed the administration of two dynamic simulator scenarios to one requalification crew; and (3) observed four evaluators administer six job performance measures, including three in the control room simulator in a dynamic mode and two in the plant under simulated conditions.

The inspectors also reviewed the remediation process for two individuals. The inspectors also reviewed the results of the annual licensed operator requalification operating examination for 2002 and 2003. The biennial written examinations that were administered in September and October 2003 were also reviewed. The results of the examinations were assessed to determine Entergy's appraisal of operator performance and the feedback of performance analysis to the requalification training program. The inspectors interviewed members of the training department and operating crews to assess the responsiveness of the licensed operator requalification program. The inspectors also observed the examination security maintenance for the operating tests during the examination week.

Additionally, the inspectors assessed the Waterford 3 plant referenced simulator for compliance with 10 CFR 55.46 using Baseline Inspection Procedure IP 71111.11 (Section 03.11). The inspectors assessed the adequacy of Entergy's simulation facility for use in operator licensing examinations and for satisfying experience requirements as prescribed in 10 CFR 55.46. The inspectors reviewed a sample of simulator performance test records (transient tests, surveillance tests, malfunction tests, and scenario-based tests), simulator work request records, and processes for ensuring simulator fidelity commensurate with 10 CFR 55.46. The inspectors also interviewed members of Entergy's simulator configuration control group as part of this review.

b. Findings

No findings of significance were identified.

.2 Quarterly Inspection

a. Inspection Scope

On October 7, 2003, the inspectors observed a licensed operator simulator training exercise. During the exercise the inspectors evaluated the operator's ability to recognize, diagnose, and respond to a steam generator tube leak followed by tube rupture. Additional challenges included loss of component cooling water Pump B, containment spray Pump B failing to start, failure of pressure level Channel X, reactor coolant Pump 2B lower seal failure, two stuck control element assemblies, and high pressure safety injection Pump A failure to start. The inspectors observed and evaluated the following areas:

- Understanding and interpreting annunciator and alarm signals
- Diagnosing events and conditions based on signals or readings
- Understanding plant systems
- Use and adherence of Technical Specifications
- Crew communications including command and control
- The crew's and evaluator's critiques

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 system that displayed performance problems:

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

- On October 3, 2003, and December 11, 2003, planned maintenance was performed on the digital fault recorders and protective relays in the 230 kV switchyard associated with offsite power to the Waterford 3 nuclear plant.

- On September 29 through September 30, 2003, during emergent repairs performed on emergency diesel generator, Train A. Repairs consisted of replacing a failed fuel line as discussed in Section 4OA3 of this report.

### 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, 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 evaluations were reviewed:

- Operability evaluation addressing dye penetrant indications on the reactor vessel head incore instruments nozzles (Condition Report CR-WF3-2003-3307)
- Operability evaluation addressing degraded shutdown cooling suction inside containment isolation valve (Condition Report CR-WF3-2003-2991)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors performed two reviews of operator workarounds. One review was performed prior to the plant being shutdown for a refueling outage that began October 20, 2003. The second review was performed immediately following the refueling outage during full power operations. The reviews evaluated the individual and cumulative effects of current operator workarounds to assess the associated impact affecting the operator’s ability to respond in a correct and timely manner to plant transients and accidents.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed 3 postmaintenance tests for selected risk-significant systems to verify their operability and functional capabilities. The inspectors considered whether testing met design and licensing bases, Technical Specifications, and licensee procedural requirements. The inspectors reviewed the testing results for the following three components:

- Nitrogen Gas Pressure Indicating Switch NG IPIS0941B following emergent repairs on September 16, 2003, due to the failure of Nitrogen Accumulator 2 Outlet Valve NG-610 to properly cycle

- Emergency diesel generator, Train A, following a failed fuel oil tubing line on September 30, 2003
- Main steam isolation Valves (MSIVs) 1 and 2 actuating system leak tests performed following system modifications in November 2003

b. Findings

Introduction. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Section XI, "Test Control," for the failure to establish adequate test controls for leak testing MSIVs 1 and 2. This performance deficiency contributed to both valves being declared inoperable due to system leaks creating a low pressure condition in the valve actuating systems.

Description. On December 6, 2003, control room operators declared MSIV 1 inoperable after identifying that the valve actuating system nitrogen pressure was less than the acceptance criteria of 2520 psig. The inspectors reviewed operator logs and noted that MSIV 2 had previously been declared inoperable for the same reason at 1:29 a.m. on November 20, 2003. Entergy had completed a refueling outage and control room operators declared MSIV 1 operable November 19, 2003, at 11:10 p.m., and MSIV 2 operable November 20, 2003, at 12:01 a.m. The inspectors reviewed work orders and noted that MSIV 1 valve actuating nitrogen system had been recharged due to low system pressure on December 1, 2003.

The inspectors reviewed the work history of the MSIV's and noted that both valves had received a modification that installed a high accuracy pressure instrument in the nitrogen actuating system. The nitrogen actuating system is a safety related system having a safety function to close each MSIV within 7 seconds following a main steam line isolation signal. The inspectors reviewed the postmaintenance test instructions and noted that the work order specified a leak test of the modified system be performed at normal system operating pressure. After reviewing operator logs and discussions with maintenance personnel that performed the leak tests, the inspectors determined that the leak tests were not performed at normal system pressure. Normal system nitrogen pressure is approximately 2600 psig and the inspectors determined that the leak tests were performed at approximately 1200 psig for MSIV 1and 930 psig for MSIV 2. The individual that performed the test did not recall the pressure that the leak tests were performed at nor did the individual recall the normal system operating pressure. The inspectors determined that Entergy had failed to establish adequate test controls for leak testing the piping connections following the modification. This resulted in the failure to identify system leaks that eventually resulted in both valves being declared inoperable.

Analysis. The deficiency associated with this finding was inadequate testing controls. The inadequate test controls failed to ensure that leak tests of the nitrogen actuating systems for the MSIV's were performed at normal system pressure (2600 psig) following modification to the systems. This performance deficiency resulted in the failure to identify system leaks which contributed to the valves being declared inoperable. This finding is more than minor because it affected the Barrier Integrity Cornerstone objective of providing reasonable assurance of the functionality of containment. The finding was evaluated using the Phase 1 significance determination process worksheet. 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, it did not result in an actual open pathway affecting the physical integrity of reactor containment, and the main steam isolation valves were inoperable for less time than the allowed Technical Specification outage time.

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 testing controls to ensure the MSIV's 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 2003-3716 and 2003-3837, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy:
NCV 50-382/2003007-01, Inadequate Test Controls of MSIV's.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

Refueling Outage 12 began on October 20, 2003, and ended on November 24, 2003. During the outage, the inspectors observed shutdown, cooldown, refueling, startup, and maintenance activities to verify that Entergy maintained the plant capabilities within the applicable Technical Specification requirements and within the scope of the outage risk plan. Specific performance activities evaluated included:

- Clearance Activities - ensured tags were properly hung and equipment appropriately configured to support the function of the clearance

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

- Reactivity Controls - ensured compliance with Technical Specifications and verified that activities, which could affect reactivity, were reviewed for proper control within the outage risk plan

- Refueling Activities - assessed compliance with Technical Specifications, verified proper tracking of fuel assemblies from the spent fuel pool to the core, and confirmed that foreign material exclusion was maintained

- Reduced Inventory and Midloop Conditions - verified that commitments to Generic Letter 88-17 were in place, that plant configuration was in accordance with those commitments, and that distractions from unexpected conditions or emergent work did not affect operator ability to maintain the required reactor vessel level

- Monitored Shutdown Cooling System - verified that operating parameters were established and maintained within the required range

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

- Spent Fuel Pool Cooling System Operation - assessed outage work for potential impact on the ability of the operations staff to operate the spent pool cooling system during and after core offload

- Containment Closure - reviewed control of containment penetrations to ensure that containment closure could be achieved within required times during various portions of the outage Reduced Inventory

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

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed or reviewed the following surveillance test to ensure the system was capable of performing its safety function and to assess its operational readiness. Specifically, the inspectors considered whether the following surveillance test met Technical Specifications, the Updated Final Safety Analysis Report, and licensee procedural requirements:

- Surveillance Procedure OP-903-026, “Emergency Core Cooling System Valve Lineup Verification,” Revision 12, performed on November 21, 2003. This surveillance verified that the appropriate valve lineup was established for low pressure safety injection system and verified the system was vented and filled with water.

b. Findings

As discussed in Section 4OA2 of this report, the inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to establish adequate corrective actions to prevent recurrence of voiding conditions affecting the operability of the low-pressure safety injection system following shutdown cooling operations.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiological Significant Areas (71121.01)

a. Inspection Scope

To review and assess Entergy's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, the inspector interviewed supervisors, radiation workers, and radiation protection personnel involved in high dose rate and high exposure jobs during the 2003 refueling outage. The inspector discussed changes to the access control program with the Radiation Protection Manager. The inspector also conducted plant walkdowns within the controlled access area and conducted independent radiation surveys of selected work areas. The following items were reviewed and compared with regulatory requirements:

- Area postings, radiation work permits (RWPs), radiological surveys, and other controls for airborne radioactivity areas, radiation areas, and high radiation areas
- High radiation area key control
- Setting, use, and response of electronic personnel dosimeter alarms
- Prejob briefings for the pressurizer heater replacement and upper guide structure movement work activities
- Conduct of work by radiation protection technicians and radiation workers in areas with the potential for high radiation dose work associated with refueling outage activities
- Dosimetry placement when work involved a significant dose gradient (primary steam generator and reactor head detensioning activities)
- Controls involved with the storage of highly radioactive items in the spent fuel pool

- Audits and self-assessments involving high radiation area controls and staff performance

- Summary of corrective action documents written since the last inspection and selected documents relating to high radiation area incidents, radiation protection technician and radiation worker errors, repetitive, and significant individual deficiencies

- Controls in place to ensure compliance with 10 CFR 20.1703(f)

There were no internal dose events which exceeded 50 millirem committed effective dose equivalent during this inspection period; therefore, this aspect to the inspection procedure could not be completed. Performance indicator reviews associated with occupational exposure control effectiveness are documented in Section 4OA1 of this report. No licensee event reports or special reports were required in this inspectable area since the previous inspection. The inspector completed all 21 of the required samples.

b. Findings

Introduction. The inspector identified a Green, noncited violation of Technical Specification 6.12.1 because Entergy failed to barricade a high radiation area to prevent inadvertent entry.

Description. On October 27, 2003, during tours of the reactor containment building the inspector noted that the high radiation area rope barricading the regenitive heat exchanger room located on the 21-foot elevation was stretched across the entrance way at a height of approximately 79 inches, which would not obstruct the entry of station workers. General area radiation levels were as high as 420 millirem per hour.

Analysis. The inspector determined that Entergy's failure to properly barricade a high radiation area as required by Technical Specification 6.12.1 is a performance deficiency. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements or licensee's procedures. The finding is greater than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute: program and process, and affected the cornerstone objective to provide adequate protection to workers health and safety from exposure to radiation. When the issue was processed through the Occupational Radiation Safety Significance Determination Process it was determined to be a "Green" finding because it was not an ALARA planning and control issue, there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.

Enforcement. Technical Specification 6.12.1. states, in part, that each high radiation area in which the intensity of radiation is greater than 100 millirem per hour but less than 1000 millirem per hour shall be barricaded.

The failure to place a high radiation barricade to obstruct entry to the regenitive heat exchanger room is a violation of Technical Specification 6.12.1. Because the finding is of very low safety significance and was entered into the corrective action program as Condition Report CR-WF3-2003-03164, this violation was treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-382/2003007-02, Failure to barricade a high radiation area.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

To assess Entergy's program to maintain occupational exposures as low as is reasonably achievable (ALARA), the inspector reviewed work activities conducted during Refueling Outage 12, and attended the pre-job ALARA brief and observed radiological work associated with the replacement of the chemical volume control system filter.

The inspector interviewed radiation protection staff members and other radiation workers to determine the level of planning, communication, ALARA practices, and supervisory oversight integrated into work planning and work activities. In addition, the following items were reviewed and compared with procedural and regulatory requirements:

- Current 3-year rolling average collective exposure
- Six ALARA prejob, in progress, and postjob reviews and associated radiation work permit packages from Refueling Outage 12 which resulted in some of the highest personnel collective exposures
- Site specific trends in collective exposures, historical data, and source-term measurements
- Site specific ALARA program procedures
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Work activity intended dose against actual dose received and the reasons for any inconsistencies
- Assumptions and basis for annual collective exposure estimates, the methodology for estimating work activity exposures, and intended dose outcomes
- Method for adjusting exposure estimates, or re-planning work, when unexpected changes in job scope or emergent work were encountered

- Use of engineering controls to achieve dose reductions and the benefits afforded by using shielding
- Historical trends and current status of tracked plant source terms and contingency plans due to changes in fuel performance or primary plant chemistry
- Radiation worker performance during work activities in radiation, high radiation or airborne radioactivity areas
- Declared pregnant workers declared during the assessment period and monitoring controls and exposure result
- Self-assessments and audits related to the ALARA program since the last inspection
- Resolution through the corrective action process of problems identified through postjob reviews and postoutage report critiques
- The effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies
- Summary of corrective action documents written since the last inspection and selected documents relating to exposure tracking, higher than planned exposure levels, radiation worker practices, repetitive, and significant individual deficiencies against the corrective action program

The inspector completed 16 sample requirements.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators listed below for the period from April 2002 through September 2003. To verify the accuracy of the performance indicator data reported during that period, performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness Performance Indicator

Licensee records reviewed included corrective action documentation that identified occurrences of locked high radiation areas (as defined in Technical Specification 6.12.2), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02). Additional documents reviewed included ALARA records and whole body counts of selected individual exposures. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspector toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. The inspector completed one of the required inspection samples.

Public Radiation Safety Cornerstone

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

Licensee documents reviewed included radiological effluent release corrective action records and annual effluent release reports during the past four quarters (no licensee event or special reports were submitted) to determine if any doses resulting from liquid or gaseous effluent releases exceeded performance indicator thresholds. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. The inspector completed one of the required inspection samples.

Barrier Integrity Cornerstone

- Unplanned Scrams

Licensee documents reviewed included control room logs, reactor power profile obtained from the plant computers, and licensee quarterly operating reports. The inspector completed one of the required inspection samples.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

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. Section 1RO8.3 evaluated the effectiveness of

Entergy's problem identification and resolution process regarding inservice inspection-related condition reports issued during the current and past refueling outages. No findings of significance were identified.

.1 Voiding in the Low Pressure Safety Injection System

a. Inspection Scope

On December 19, 2003, the inspectors completed a review of Entergy's actions regarding voiding in the low pressure safety injection system. Entergy's previous actions to address voiding conditions affecting the emergency core cooling systems for a number of years are documented in NRC Inspection Report 05000382/2002005. The long-term voiding issues have contributed to equipment failures along with operations, engineering, and radiological challenges. The inspectors conducted interviews with responsible engineers, operators, and managers and reviewed relevant documents and drawings.

b. Findings

Introduction. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to establish adequate corrective actions to prevent recurrence of voiding conditions affecting the operability of the low pressure safety injection system following shutdown cooling operations.

Description. On November 21, 2003, following restart from Refueling Outage 12, Entergy performed ultrasonic testing on low pressure safety injection system, Trains A and B, per Procedure OP-903-026, "Emergency Core Cooling System Valve Lineup Verification," Revision 12, to determine if the piping was water solid. The ultrasonic testing identified gas voids at Containment Penetration 38 and Vent Valve SI-134A that resulted in the low pressure safety injection system Train A being inoperable in accordance with the procedural acceptance criteria. Subsequently, ultrasonic testing identified gas voids in the low pressure safety injection system, Train B, at Vent Valves SI-133B and SI-134B and at Containment Penetrations 36 and 37. The gas void at Vent Valve SI-134B rendered low pressure safety injection system Train B inoperable.

The inspectors noted that one day following Refueling Outage 11, a gas void was identified in low pressure safety injection system Train B, which required Train B to be declared inoperable (Documented in NRC Inspection Reports 05000382/2002002 and 05000-382/2002005). Entergy determined that the root cause for gas voiding in the low pressure safety injection system was an inadequate plan to vent the gas following the low pressure safety injection system realignment from shutdown cooling to the safety injection mode. Entergy placed this degraded condition into their corrective action process as Condition Report 2002-00818. Corrective actions assigned were to vent and sweep the low pressure safety injection system with less gas saturated water from the refueling water storage pool. Additionally, procedural guidance requiring operators to ensure work orders were generated to vent and fill the low pressure safety injection system following shutdown cooling operations was implemented.

Following the identification of gas voids on November 21, 2003, engineering and operations personnel stated that although work orders were written, operations failed to accomplish the vent and fill tasks in a timely manner resulting in the accumulation of voids that rendered the systems inoperable per the proceduralized acceptance criteria. The inspectors determined that the failure to effectively implement the corrective actions directed by Condition Report 2002-00818 resulted in unacceptable voids in the low pressure safety injection system. This failure resulted in a recurrence of a significant condition adverse to quality and was determined to be a violation of 10 CFR Part 50, Appendix B, “Corrective Action.”

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 unacceptable gas voids identified in the low pressure safety injection system following Refueling Outage 11 were not effectively implemented and failed to prevent recurrence following Refueling Outage 12. This finding is greater than minor because it affected the mitigating system objective to ensure the reliability and availability of the low pressure safety injection system to respond to an initiating event. The problem if left uncorrected would become a more significant safety concern. The significance of this finding was determined to be of very low safety significance because Train B was inoperable for less than the Technical Specification allowed outage time and Train A was determined to be degraded but operable in accordance with Generic Letter 91-18 guidance.

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 unacceptable void accumulations in the low pressure safety injection system following shutdown cooling operations is a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action." Because this finding is of very low safety significance and has been entered into Entergy’s corrective action program as Condition Reports 2003-3740, -3858, and -3901, this violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000382/2003007-03, Ineffective Corrective Actions to Prevent Recurrence of Voiding Conditions.

.2 Alloy 600 Nozzle Cracking

a. Inspection Scope

During refueling Outage 12 (October 20, 2003, through November 24, 2003) Entergy identified pressure boundary leakage emanating from three Alloy 600 reactor coolant system nozzles. The inspectors reviewed Entergy's corrective actions associated with the multiple nozzle failures. Additionally, the inspectors reviewed the corrective and preventive maintenance history of the reactor coolant system Alloy 600 nozzles. The inspectors also reviewed previous corrective actions addressing pressure boundary leakage to evaluate their effectiveness in preventing recurrence.

b. Findings

Introduction. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to implement effective corrective actions resulting in recurrences of pressure boundary leakage due to primary water stress corrosion cracking of Alloy 600 reactor coolant system nozzles.

Discussion. Entergy identified three reactor coolant system Alloy 600 nozzle leaks during Refueling Outage 12. The nozzles affected were the reactor coolant system hot Leg 2 Instrument Nozzle RC-IPT-0106B and two pressurizer heater sleeve Nozzles C-1 and C-3. Each nozzle leak was determined by Entergy to be the result of primary water stress corrosion cracking. Each nozzle leak was repaired prior to the end of the refueling outage.

The inspectors reviewed the history of the Alloy 600 nozzles at Waterford 3 and noted that multiple failures had previously occurred due to primary water stress corrosion cracking. In Refueling Outage 9, three hot leg nozzles and two pressurizer top nozzles were found leaking. In Refueling Outage 10, one pressurizer heater sleeve nozzle was found leaking. The inspectors reviewed Condition Reports 1999-00204, 1999-00232, 1999-00234, 2000-1250, 2003-03130, and 2003-03110. Review of these corrective action documents revealed that no replacement plans had been established by Entergy to repair or replace the Alloy 600 nozzles throughout the reactor coolant system that were susceptible to primary water stress corrosion cracking. The inspectors noted that Condition Report 2000-1250 stated, "Due to the nature of primary water stress corrosion cracking and the use of Inconel 600 at Waterford 3, recurrence of similar leaks is considered beyond Waterford 3 control." The inspectors noted that with the current materials and ongoing actions, that Waterford 3 would be susceptible to future pressure boundary leakage caused by primary water stress corrosion cracking of Alloy 600 nozzle material. Operation of Waterford 3 with reactor coolant system boundary leakage is a condition prohibited by plant Technical Specification 3.4.5.2.a during Modes 1, 2, 3, and 4.

The inspectors noted that Entergy had not initiated actions to prevent the occurrence of pressure boundary leakage, through the Inconel 600 material nozzles, during either

Refueling Outages 11 or 12. The inspectors also noted that no inspections other than visual examinations to find leakage were being performed by Entergy to detect degradation of the nozzles that would allow for repairs or replacement prior to reactor coolant system pressure boundary leakage occurring. The inspectors determined that Entergy had not established adequate measures to prevent recurrence of reactor coolant system pressure boundary leakage, due to primary water stress corrosion cracking of Alloy 600 nozzles, a 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, Entergy had not established corrective measures to preclude multiple occurrences of reactor coolant system pressure boundary leakage, during an operating cycle, due to primary water stress corrosion cracking of Alloy 600 nozzle material. This finding was greater than minor because it affected the reactor safety barrier integrity cornerstone objective for providing reasonable assurance that the physical design barriers protect the public from radionuclide releases caused by accidents or events. Using NRC Manual Chapter 0609 significance determination process Phase 1 Screening Worksheet this performance deficiency affected the reactor coolant system barrier function requiring a Phase 2 analysis. The results of the Phase 2 and 3 analysis determined that this finding was of very low safety significance based on the cracks being axial in nature (does not contribute substantially to a loss of coolant accident) and the leaks resulted in a buildup of only minor boric acid residue indicative of only trace amounts of through wall leakage. The leak rates identified were well within the capacity of a single charging pump.

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 reactor coolant system pressure boundary leakage due to primary water stress corrosion cracking of Alloy 600 nozzle material is considered a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action." Because this finding is of very low safety significance and has been entered into Entergy's corrective action program as Condition Reports 2003-03130 and 2003-03110, this violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000382/2003007-04, Ineffective Corrective Actions to Prevent Recurrence of PWSCC of Alloy 600 material.

4OA3 Event Followup (71153)

.1 Failure of the Train A Emergency Diesel Generator

Description of Event

On September 29, 2003, during the performance of a monthly surveillance run, the Train A emergency diesel generator experienced a fuel line failure. Approximately 3 hours into the surveillance an operator in the Train A diesel room observed the fuel line break and immediately shutdown the diesel locally in approximately 15 seconds. The operator reported seeing a solid stream of fuel oil being discharged from the fuel line break located on the left cylinder bank side of the diesel generator. Approximately 70 gallons of fuel oil was discharged from the line break. Waterford personnel performed a field inspection and identified that the 3/4 inch stainless steel fuel supply tube had sheared 360 degrees where the tube inserted into a Swagelok compression fitting.

Entergy assembled a root cause analysis team to investigate the cause of the failure and develop a corrective action plan. Plant personnel replaced the failed tubing, retested the Train A emergency diesel, and restored the diesel to operable status on September 30, 2003. Examination of the failed tubing indicated that fatigue failure resulted in the break where the tip of the back ferrule of the compression fitting contacted the tubing. Entergy determined the Train B emergency diesel generator was not susceptible to the same failure mechanism since the fuel lines were the original lines having never been replaced by Entergy.

a. Inspection Scope

The inspector reviewed the sequence of events related to the emergency diesel generator fuel oil line failure.

The inspector assessed Entergy's immediate actions and subsequent evaluation of the Train A emergency diesel generator failure that occurred on September 29, 2003.

The inspector evaluated pertinent industry operating experience and potential precursors to the failure of emergency diesel generator fuel oil line at the Swagelock fitting.

The inspector reviewed and assessed Entergy's corrective actions to verify that they have adequately evaluated and addressed the extent of condition including generic implications.

The inspector reviewed Entergy's root cause evaluation determination for independence, completeness, and accuracy.

The inspector, along with a senior reactor analyst inspector, assessed the safety significance associated with the Train A emergency diesel generator failure.

b. Findings

Introduction. A self-revealing apparent violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified for the failure to establish appropriate instructions and accomplish those instructions for proper installation of a fuel line for Train A emergency diesel generator in May of 2003. This failure resulted in uneven and excessive scoring of the tubing that ultimately led to a complete 360 degree failure of the fuel supply line on September 29, 2003, during a monthly surveillance test.

Description. In May of 2003, Entergy performed an overhaul on the Train A emergency diesel generator. During this overhaul the fuel oil header left/right bank cross connect tubing and associated fittings were replaced to repair a small fuel oil leak that had been discovered previously. The tubing was 316 grade stainless steel, 3/4 inch outer diameter, with a nominal wall thickness of 0.049 inches. The tubing was bent by mechanical craft personnel at four locations and spanned approximately 5 feet. The replacement compression fittings were manufactured by Swagelok. After approximately 28.7 hours of runtime, following the May overhaul, failure of the replaced fuel oil line occurred on September 29, 2003 during a monthly surveillance test. Entergy noted that a complete 360 degree failure of the tubing occurred at the compression fitting that attached the fuel line to the diesel engine left cylinder bank.

Entergy sent the failed specimen to two laboratories for examination to determine the root cause of the failure. Both labs concluded that fatigue failure of the tubing occurred at the point where the back ferrule of the compression fitting contacted the outer tubing surface. It was noted that the tube failure was along the front edge of the back ferrule and that the outer circumference of the tubing along the fracture was unevenly scored, up to 30 percent of the tubing thickness. According to Swagelok, a correct installation would result in an evenly scored tube, approximately 10 percent of the tubing thickness. Entergy determined that improper alignment of the tubing in the compression fitting and potential over tightening of the compression fitting resulted in the uneven and excessive scoring. With these conditions established, the vibrational stresses subjected to the flawed tubing connection experienced during operation of the diesel generator resulted in fatigue failure of the tubing on September 29, 2003.

The inspectors reviewed Entergy’s analysis of the event contained in document CR-WF3-2003-02759. Entergy concluded that the installation of the replacement tubing offered minimal margin for error when considering the following design attributes:

- Specified material is thin walled (0.049 inches)
- Large bore tubing ≥ ½ inch
- Configuration is complex containing multiple tube bends

- Swagelok fittings produce tube scoring
- Vibration is present

Entergy determined that all these factors played a role in causing the tubing failure when coupled with the tubing not being correctly installed into the Swagelok compression fitting. Entergy determined that the extent of condition for this type of failure mechanism was isolated to the Train A emergency diesel generator. A review of past events revealed that there had been isolated tubing leaks or failures located at Swagelok fittings that had all occurred greater than three years previous to the failure on September 29, 2003. The inspectors reviewed these instances and found that there were slight differences between the identified failure mechanisms, but did note that one common corrective action was to provide additional training to mechanical maintenance personnel on how to appropriately install a Swagelok compression fitting application. The inspectors noted that Entergy relied on skill of the craft to install the fittings with no detailed instructions or quality control checks provided to ensure the fittings were made correctly. The inspectors noted that the Swagelok manual contained detailed instructions for installing the fitting and also recommended the use of a “depth marking tool” that could be used to ensure proper tube alignment within the compression fitting. In review of the training materials and discussions with maintenance department personnel the inspectors noted that not all Swagelok recommended installation practices were being implemented by Entergy, including use of the depth marking tool. The inspectors discussed these observations with licensee senior management who indicated that they were evaluating enhancements that included more detailed instructions.

Analysis. The deficiency associated with this event was the failure to establish appropriate measures to ensure proper installation of a replacement fuel oil line on the Train A emergency diesel generator in May of 2003. This failure resulted in uneven and excessive scoring of the tubing that ultimately led to a complete 360 degree failure of the fuel supply line on September 29, 2003. The finding was greater than minor because it directly impacted the availability and reliability of an emergency diesel generator which is used to mitigate the loss of AC power to the respective safety related bus. The finding was determined to be potentially greater than Green based on a Phase 1, Phase 2, and Phase 3 analysis.

Significance determination process Phase 1:

In accordance with NRC Inspection Manual Chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the inspectors conducted a significance determination Phase 1 screening and determined that the finding resulted in loss of the safety function of the Train A emergency diesel generator for greater than the Technical Specification allowed outage time. Therefore, a Significance determination process Phase 2 evaluation was required.

Significance determination process Phase 2:

The Risk-Informed Inspection Notebook for Waterford Nuclear Plant Unit 3, Revision 1, September 2, 2003, was utilized for the Phase 2 evaluation of the inoperable Train A emergency diesel generator. The following steps and the associated findings are listed below:

- Select or define the applicable initiating event scenarios:

  Table 2, “Initiators and System Dependency for Waterford Nuclear Plant, Unit 3,” was reviewed to determine that the loss of offsite power (LOOP) initiating event scenario was the only scenario that needed to be analyzed due to the failure of the Train A emergency diesel generator.

- Estimate the likelihood of scenario initiating events and conditions:

  The performance deficiency was assumed to exist for greater than 30 days based on the failure of the diesel generator being expected to occur within its mission time. The mechanism leading to the fuel line break was fatigue failure caused by vibration that only occurred while the engine was running. No degradation was assumed to occur while the engine was idle. Therefore, the diesel was destined to fail approximately 28 hours after the fuel line was replaced in May of 2003. Using Table 1, “Categories of Initiating Events for Waterford Nuclear Plant, Unit 3,” the initiating event likelihood for loss of offsite power was determined to be valued at 2.

- Estimate the remaining mitigation capability:

  Using the significance determination process worksheet for loss of offsite power (Table 3.6, SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 - Loss of Offsite Power (LOOP)), Sequences 1, 2, and 3, the following results were assigned for each:

  Sequence 1: LOOP-EFW - 6
  Sequence 2: LOOP-EDG-REC8 - 6
  Sequence 3: LOOP-EDG-TDEFW-REC1 - 6

  Estimate the risk significance of the inspection finding:

  NRC Inspection Manual Chapter 0609, “Significance Determination Process,” Appendix A, Attachment 1, “Counting Rule Worksheet,” was utilized using three sequences that resulted in values of 6. Since Step 10 was greater than zero, the risk significance of the inspection finding was determined to be at low to moderate safety significance (White).

As a result of the White finding in the Phase 2 evaluation, a Phase 3 evaluation was performed.

Significance Determination Process Phase 3 Analysis

The following table presents the running history of the "A" emergency diesel generator from the time of the maintenance until the run failure.

| Date | Event | Run Time (PMT) | Run Time (Surveillance) | Total Run Time for Day |
|---|---|---|---|---|
| May 16-17, 2003 | EDG A Maintenance Outage | 39 min | 4:15 | 4:54 |
| June 9, 2003 | Monthly Surveillance | | 4:46 | 4:46 |
| July 8, 2003 | Monthly Surveillance | | 5:23 | 5:23 |
| August 4, 2003 | Monthly Surveillance | | 4:36 | 4:36 |
| September 2, 2003 | PMT | 20 | | 0:20 |
| September 2, 2003 | Monthly Surveillance | | 5:02 | 5:02 |
| September 29, 2003 | Monthly Surveillance | | 2:48 (fuel oil line break) | 2:48 |
| | Cumulative run time since tubing replaced. | | | 27 hours 49 minutes total run time for EDG A |

Assumptions:

- The mechanism leading to the fuel line leak was fatigue failure caused by vibration that occurred while the engine was running. No degradation occurred while the engine was idle. Therefore, the diesel was destined to fail after 28 hours of run time, regardless of how this time was accrued.

- The primary period of risk was the 27 days of standby service while the EDG had only 2 hours, 48 minutes of run time remaining. Prior to this period, the EDG had approximately 8 hours or more of run time remaining, and a failure after 8 hours of accident recovery would have been much less important because of the higher probability of recovering offsite power or the Train B EDG. The analyst evaluated the 27 day high-risk period and applied an adjustment factor based on information received from Entergy to account for the other periods of exposure.

- Based on information received from Entergy, a nonrecovery probability of 0.6 was applied to each cutset that contained a Train B EDG fail-to-start basic event, but no recovery was assumed for fail-to-run or test/maintenance situations.

- Based on information received from Entergy, a nonrecovery probability of 0.1 was applied for the Train A EDG, after its failure from a fuel line failure. This assumption was based on a statistical analysis performed by Entergy and a walk-through simulation, where a maintenance technician procured the necessary tools and manufactured a replacement fuel line segment in approximately 25 minutes. Although this assumption was used in this analysis, the analyst recognized that certain factors such as stress, lack of procedures, diversion to other tasks, having only emergency lighting, and the presence of excess fuel oil could make the fuel line repair more likely to fail than once in every 10 attempts. A sensitivity analysis was performed that did not provide recovery credit.

- The postprocessing application of the non-recovery probabilities for EDG A and B were applied only to SBO sequences 2 and 13 in accordance with the technical advice of Idaho National Engineering and Environmental Laboratory (INEEL). Other sequences included recoveries inherent to the model. SBO sequences 2 and 13 comprised approximately 80 percent of the change in risk.

- The failure mechanism on EDG A was not a susceptible failure for EDG B as noted above, therefore, to prevent the SPAR model from imputing a higher failure rate for EDG "B," the basic event for EDG "A" fail-to-run was set to a probability of 1.0 in lieu of setting it to "TRUE."

- Because EDG A would have run for 2 hours and 48 minutes during the final 27 days of the exposure period, the analyst changed the LOOP initiating event frequency in both the base and evaluation cases to reflect the probability that offsite power would be restored prior to the diesel failure. This is based on the first-order assumption that if offsite power is recovered prior to the diesel failure, the recovery will be successful. Using the NUREG-5496 for Waterford 3, the frequency weighted average probability of recovering offsite power in 2 hours, 48 minutes is

80 percent. Therefore, the LOOP frequency was changed from 5.2E-6/hr. to 1.04E-6/hr. Inherent to the analysis was the bounding assumption that if EDG B fails, it will do so before the failure of EDG A. To be consistent, the basic event OEP-XHE-NOREC-ST (Operator fails to recover offsite power in the short term) was set to a probability of 1.0 in both the base and evaluation cases, since this is implied in the adjustment to the loss of offsite power frequency.

Quantification of the Change in Risk:

The analyst used SAPHIRE 6.79 software and the Waterford SPAR, Revision 3i model, further revised by INEEL to include updated offsite power recovery curves and reactor coolant pump seal failure probabilities.

The update was accomplished to make LOOP recovery times consistent with NUREG-5496, which reported generally longer times of offsite power recovery than previous studies. As a protocol, the mission time assigned to the emergency diesel generator was made equal to the time after a LOOP needed to achieve a 95 percent probability of recovering offsite power. As a consequence, the mission time assigned to the emergency diesel generators was extended to 15 hours, making the fail-to-run emergency diesel generator events more important in the calculation. The NRC recently used a similar update to evaluate an event at the Salem plant in Region I.

To update the base model the following change sets were inserted:

- IE LOOP set to a probability of 1.04E-6/hr
- OEP-XHE-NOREC-ST set to a probability of 1.0

The result obtained was 6.84E-9/hr.

This result was further adjusted to account for the 0.6 non-recovery probability of the EDG B fail-to-start events in SBO sequences 2 and 13. The total CDF of these cutsets was 1.44E-10/hr. Therefore the adjusted base model result is:

- 6.84E-9/hr. - 1.44E-10/yr. (1 - 0.6) = 6.78E-9/hr.

To evaluate the risk associated with the performance deficiency, the following change sets were applied:

- IE LOOP set to a probability of 1.04E-6/hr
- OEP-XHE-NOREC-ST set to a probability of 1.0
- EPS-DGN-FR-DG3A (Diesel Generator 3A-S Fails to Run) set to a probability of 1.0

The result obtained was 2.62E-8/hr.

Adjustments were made to cutsets containing SBO sequences 2 and 13 with either an EDG A FTR or EDG B FTS events, as follows:

SBO sequences 2 and 13 exclusively contain the basic event OEP-XHE-NOREC-BD (Operators fail to recover AC power before battery depletion)

The recovery of both EDGs applies to the total CDF of cutsets containing OEP-XHE-NOREC-BD and EPS-DGN-FS-DG3B (EDG B fails to start) and EPS-DGN-FR-DG3A (Diesel Generator 3A-S Fails to Run) =1.23E-9/hr. Application of 0.1 non-recovery probability for EDG A and 0.6 non-recovery probability for EDG B results in a total CDF <u>reduction</u> of 1.23E-9/hr. (1.0 - (0.1) (0.6)) = 1.16E-9/hr.

The recovery of EDG A only applies to the total CDF of cutsets that contain OEP-XHE-NOREC-BD and EPS-DGN-FR-DG3A (EDG A fails to run), excluding the cutsets in the group above that contain EPS-DGN-FS-DG3B.

The total CDF of this group is 1.48E-8. Application of 0.1 non-recovery probability for EDG A results in a total CDF <u>reduction</u> of 1.48E-8/hr. (1.0 - 0.1) = 1.33E-8/hr.

Therefore, the revised CDF for the evaluation case is:

- 2.62E-8/hr. - 1.16E-9/hr. - 1.33E-8/hr. = 1.17E-8/hr.

The change in frequency attributable to the performance deficiency is:

- 1.17E-8/hr. - 6.78E-9hr. = 4.92E-9/hr.

The exposure period of 27 days consists of 648 hours. Therefore the delta CDF of the performance deficiency is:

- 4.92E-9/hr. (648 hours/yr.) = 3.19E-6/yr.

<u>Consideration of other periods of exposure</u>

The analyst evaluated only the final 27 days of exposure when EDG A had only approximately three hours of run time remaining. Some risk was also incurred when the EDG had additional run time remaining. In Entergy's evaluation the percentage of the total calculated risk associated with the final 27-day exposure period was 61.2%. The analyst adjusted the CDF result obtained above by this percentage to approximate the risk incurred during the entire exposure period.

Adjusted CDF = 3.19E-6/yr (1/0.612) = 5.21E-6/yr

<u>External Initiators</u>:

The analyst concluded that external initiators would have a negligible impact on the final result.

High winds or other weather conditions that could cause a loss of offsite power were included in the updated database used to estimate the LOOP frequency, and equipment important to mitigating the consequences of these occurrences are adequately protected from these events.

Seismic events at the plant are rare and of low magnitude, and the plant is isolated by being situated on a "floating island." A seismic event is not likely to cause an earlier failure of the fuel line fitting because the entire skid would move as a unit and would not be subjected to the differential stresses caused by the vibrations of a running engine.

Internal flooding would not likely cause a loss of offsite power or failure of the Train B EDG and would therefore have little impact on the analysis.

Fires that could cause a loss of offsite power were isolated to two fire areas, both with frequencies more than two orders of magnitude less than the LOOP frequency. A fire initiating in the Train A EDG room as a consequence of the as-found fuel spill was very unlikely because of the lack of sufficient ignition temperatures on surfaces exposed to the spill. Additionally, a fire in this room would only affect the operation of the Train A EDG and would not impede operator access to other mitigating system components. The room contained automatic detection and suppression devices.

In summary, the overall effect of external initiators would be very small compared to the internal result.

<u>Large early release frequency</u>:

The analyst reviewed the finding for impact on large early release using Inspection Manual Chapter 0609, Appendix H. On a loss of power, all containment isolation and purge valves fail closed. There existed no other conditions involving containment integrity. Therefore LERF, though slightly increased because of the increase in delta-CDF, was well below the E-7/yr. threshold. This is because a release would have occurred only in the event of a concurrent containment boundary failure.

Sensitivity Considerations:

If recovery of EDG A is not credited, the resulting delta CDF is 2.00E-5/yr. If recovery is not credited for either EDG, the result is 2.05E-5/yr.

Conclusion:

The condition was of low to moderate safety significance (WHITE). If credit is not applied for recovery of EDG A, the result is one of substantial safety significance (YELLOW).

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states in part, that "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." The failure to establish appropriate instructions and accomplish those instructions for installation of the fuel line for Train A emergency diesel generator resulting in the fuel line failure on September 29, 2003, is a violation of 10 CFR Part 50, Appendix B, Criterion V. Pending determination of the finding's final safety significance, this finding is identified as Apparent Violation (AV) 05000382/2003007-05, Failure to establish appropriate instructions and implement those instructions.

.2 (Closed) Licensee Event Report 50-382/2003-001-00: Loose Breaker Fuse Rendered One Bank of Pressurizer Proportional Heaters Inoperable

On July 24, 2003, Entergy identified that a loose control power fuse for the Control Element Drive Motor Generator Set B breaker rendered one bank of pressurizer proportional heaters inoperable beyond the allowed outage time of Technical Specification 3.4.3.1. This was determined to be a violation of Technical Specification 3.4.3.1 (See section 40A7 for details). This finding is more than minor because it had a credible impact on safety, in that if the redundant group of proportional heaters did not function, reactor coolant system pressure control under natural circulation conditions could not be ensured. This finding affects the Mitigating Systems Cornerstone. Using the significance determination process, this issue was determined to have a very low safety significance, since only one train of proportional heaters is required to control reactor coolant system pressure under natural circulation conditions and operators could manually align the heaters to their emergency power source locally had the automatic transfer failed during a loss of normal power event. This issue was entered into Entergy's corrective action process as Condition Report CR-WF3-3003-2076.

.3 (Closed) Licensee Event Report 50-382/2003-003-00: Reactor Coolant System Boundary Leakage Due to Primary Water Stress Corrosion Cracking

During Refueling Outage 12, Entergy identified three indications of reactor coolant system pressure boundary leakage. The first indication was identified on October 24, 2003, on the hot leg #2 instrument nozzle connected to instrument RC-IPT-0106B. The other two indications of leakage were identified on October 26, 2003. These indications were identified on pressurizer heater sleeves C-1 and C-3. This issue is addressed in Section 4OA2.2 of this report.

4OA4 Crosscutting Aspects of Findings

Section 1R19 of the report describes a human performance crosscutting issue where personnel failed to establish appropriate postmaintenance testing criteria following a modification to the main steam isolation valve nitrogen actuating system.

Section 4OA3 of the report describes a human performance crosscutting issue where maintenance personnel performed improper installation of the EDG Train A fuel oil line.

4OA5 Other Activities

.1 Reactor Pressure Vessel Head and Vessel Head penetration Nozzles (Temporary Instruction 2515/150, Revision 2)

a. Inspection Scope

The inspectors verified that Entergy's susceptibility ranking was "high" based on the calculated effective degradation years being 16.95 years through Cycle 12. Entergy used plant specific temperature data in their susceptibility ranking calculation.

The inspectors noted that examinations were performed by contract Westinghouse and Entergy personnel. Contract personnel had been qualified using licensee qualification procedures and all personnel had been qualified using procedures that satisfied applicable requirements of SNT-TC-1A and ASME Section XI. Westinghouse personnel performed eddy current testing and ultrasonic examinations. Entergy personnel performed dye penetrant testing.

The reactor vessel had 102 penetrations (1 reactor head vent, 10 incore instrument nozzles, and 91 control element drive mechanisms). Entergy performed dye penetrant, ultrasonic, and eddy current examinations on the penetrations to identify flaws. The reactor head vent was analyzed using eddy current testing. The control element drive mechanisms were analyzed using ultrasonic testing. The incore instruments were analyzed using a combination of eddy current testing, ultrasonic testing, and dye penetrant testing. Entergy also performed a bare metal visual inspection of 83 vessel head penetrations. A bare metal visual inspection of the remaining 19 could not be performed due to concerns with damaging the head vent line. However, Entergy did perform a complete inspection of the reactor vessel head using a boroscope. The inspectors observed the accessible areas of the vessel head and observed selected portions of the videotaped results of the boroscope data. No evidence of boric acid leakage was noted.

The inspectors reviewed the results of the eddy current testing of the reactor head vent, the results of 9 ultrasonic tests on control element drive mechanism, and results of eddy current, dye penetrant, and ultrasonic tests of 2 incore instruments nozzles. All the examinations were performed in accordance with approved procedures. The inspectors reviewed testing results for incore Instrument Penetrations 94 and 98. No indications were identified on incore Instrument Penetration 98. However, dye penetrant examinations did identify a ½-inch rounded indication at the nozzle to toe weld of Instrument Penetration 94. This indication exceeded the code criteria for allowable indication size. This indication was removed mechanically. Additional indications were also identified on incore Instrument Penetrations 92 and 93. Indications on Instrument Penetration 92 were rounded 3/32-inch indications on the nozzle to weld toe. The indication on Instrument Penetration 93 was a 3/16-inch linear indication at the nozzle to weld toe. This indication exceeded the code criteria for allowable indication size. The indications were also removed by mechanical removal. The indications were believed to be weld flaws. No evidence of leakage was found. Followup examinations after repair revealed no relevant indications. Entergy initiated Condition Report 2003-3307 based on the results of the of the dye penetrant examinations.

The inspectors reviewed eddy current test results of the reactor head vent penetration and control element driven mechanism (CEDM) Penetrations 18, 21, 30, 44, 52, 53, 56, 66, and 87. No indications or evidence of leakage were identified.

Entergy used ultrasonic examinations data to provide an assessment of leakage into the interference fit zone. Guidance for performing this assessment was contained in Procedure WDI-UT-013, “CRDM/ICI UT Analysis Guidelines,” Revision 3.

Entergy also performed visual inspections of the top of the cooling shroud. These inspections were performed using a video camera and a boroscope. No indications of boric acid buildup were noted.

The inspectors reviewed the approved relaxation requests from the NRC Order EA-03-009, "Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors."

- Relaxation request dated July 1, 2003, allowed Entergy to use eddy current testing to inspect the vent line nozzle and J-groove weld instead of ultrasonic testing. The relaxation was approved on October 2, 2003.

- Relaxation request dated September 15, 2003, allowed the control element drive mechanism nozzles to be inspected using a three-step alternative involving an analysis technique, ultrasonic testing, and augmented surface examination. This request was approved November 12, 2003.

- Relaxation request dated October 24, 2003, allowed ultrasonic testing and surface examinations of the incore instrument nozzles. This relaxation request was approved on November 7, 2003.

b. Findings

No findings of significance were identified.

.2 Reactor Containment Sump Blockage (Temporary Instruction 2515/153, Revision 0)

a. Inspection Scope

On November 17, 2003, the inspectors completed a detailed walkdown of the safety injection sump, drainage paths to the safety injection sump, and evaluated insulation and material coatings used in containment that could contribute to sump blockage in a postaccident scenario. The inspectors verified that the safety injection sump screen was free of adverse gaps and breaches to prevent debris from entering the safety injection system suction piping. The inspectors assessed Entergy's containment foreign material management control program and verified that Entergy maintained adequate cleanliness standards to prevent debris transport that could lead to potential blockage of the safety injection sump's screens. The safety injection sump design and the containment drainage arrangement was assessed using applicable sections of the Updated Final Safety Analysis Report, contractor test modeling of the safety injection system sump and interviews with civil, mechanical, and system engineers. The inspectors will complete the inspection of Entergy's compensatory measures in response to degraded containment sump performance following development of training and procedures scheduled to be completed in April 2004.

b. Findings

No findings of significance were identified.

.3 (Closed URI 05000382/0310-01): Possibility of flooding both emergency diesel generator fuel oil storage tank rooms in the event of a flood and subsequent loss of offsite power.

a. Inspection Scope

As discussed in NRC Inspection Report 05000382/2003010 a potential finding was identified in that both emergency diesel generators could be lost due to potential flooding in the emergency diesel generator fuel oil storage tank rooms due to leaking check valves installed in the industrial waste nonsafety-related drain systems connected to the rooms. The inspectors reviewed Entergy's analysis of this potential condition and discussed the results of the analysis with the responsible system engineers.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

On December 5, 2003, the inspector presented the ALARA Planning and Controls inspection results to Mr. J. Venable, Site Vice-President and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection. The inspector asked Entergy whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

The inspectors presented the results of the inservice inspection effort to Mr. R. Fili, Engineering Program Manager and other members of licensee management on November 6, 2003. Entergy management acknowledged the inspection findings. The inspectors asked Entergy whether any materials examined during the inspection should be considered proprietary. Several documents were proprietary information as identified by Entergy. The inspectors informed Entergy that these documents would be destroyed upon completion of their review.

The resident inspectors presented the integrated inspection results to Mr. J. Venable, Site Vice-President and other members of Entergy management at the conclusion of the inspection on January 5, 2004. Entergy acknowledged the findings presented. The inspectors asked Entergy 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 Entergy and is a violation of NRC requirements, which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

Technical Specification 3.4.3.1 requires at least two groups of pressurizer proportional heaters be operable. With one group of pressurizer proportional heaters inoperable, Entergy must restore the other group within 72 hours or be in Mode 3 within 6 hours. The proportional heaters remained inoperable for about 4 days while the unit was in Mode 1. Entergy had failed to meet Technical Specification requirements. This issue was determined to be more than minor because pressurizer proportional heaters help to ensure the capability of systems that respond to initiating events and was of very low safety significant because only one train of proportional heaters is required to control reactor coolant system pressure under natural circulation conditions and operators could manually align the heaters to their emergency power source locally had the automatic transfer failed during a loss of normal power event. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This issue was entered into Entergy's corrective action process as Condition Report CR-WF3-3003-2076.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee Personnel

S. Anders, Superintendent, Plant Security
T. Brumfield, Manager Quality Assurance
L. Dauzat, Supervisor, Radiation Protection
J. R. Douet, General Manager, Plant Operations
R. Fili, Engineering Program Manager
R. Fletcher, Operations Training Supervisor
C. Fugate, Assistant Manager, Operations
T. Gaudet, Director, Planning and Scheduling
B. Greeson, Code Program Supervisor, Arkansas Nuclear One
B. Houston, Manager, Radiation Protection
R. Jones, Simulator Support Supervisor
P. Kelly, Supervisor, Radiation Protection
C. Lambert, Director, Engineering
J. Laque, Manager, Maintenance
R. Murillo, Engineer, Licensing
R. Osborne, Manager, System Engineering
W. H. Pendergrass, Assistant Operations Manager (Support)
K. Peters, Director, Nuclear Safety Assurance/Emergency Preparedness
G. Pierce, Chemistry Superintendent
G. Scott, Engineer, Licensing
G. Sen, Manager, Licensing
T. E. Tankersley, Manager, Training
J. Venable, Vice President, Operations
K. T. Walsh, Manager, Operations
D. Weber, Codes Program Steam Generator Engineer

NRC

V. Gaddy, Senior Project Engineer, Region IV

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

| | | |
|---|---|---|
| 05000382/2003007-01 | NCV | Inadequate Test Controls of MSIV's (Section 1R19) |
| 05000382/2003007-02 | NCV | Failure to barricade a high radiation area (Section 2OS1) |
| 05000382/2003007-03 | NCV | Ineffective Corrective Actions to Prevent Recurrence of Voiding Conditions (Section 4OA2.1) |
| 05000382/2003007-04 | NCV | Ineffective Corrective Actions to Prevent Recurrence of PWSCC of Alloy 600 material (Section 4OA2.2) |

| | | |
|---|---|---|
| 05000382/2003007-05 | AV | Failure to establish appropriate instructions and implement those instructions (Section 4OA3.1) |

Closed

| | | |
|---|---|---|
| 05000382/2003007-01 | NCV | Inadequate Test Controls of MSIV's (Section 1R19) |
| 05000382/2003007-02 | NCV | Failure to barricade a high radiation area (Section 2OS1) |
| 05000382/2003007-03 | NCV | Ineffective Corrective Actions to Prevent Recurrence of Voiding Conditions (Section 4OA2.1) |
| 05000382/2003007-04 | NCV | Ineffective Corrective Actions to Prevent Recurrence of PWSCC of Alloy 600 material (Section 4OA2.2) |
| 05000382/2003010-01 | URI | Possibility of flooding both emergency diesel generator fuel oil storage tank rooms in the event of a flood and subsequent loss of offsite power (Section 4OA5.3) |
| 05000382/2003-001-00 | LER | Loose Breaker Fuse Rendered One Bank of Pressurizer Proportional Heaters Inoperable (Section 4OA3.2) |
| 05000382/2003-003-00 | LER | Reactor Coolant System Pressure Boundary Leakage Due to Primary Water Stress Corrosion Cracking (Section 4.OA3.3) |

## LIST OF DOCUMENTS REVIEWED

**IP 71111.09**

Procedures

DC-317, "Entergy Steam Generator Administrative Procedure," Revision 1
NOECP-252, "Steam Generator Eddy Current Inservice Testing," Revision 8
NOECP-257, "Steam Generator Secondary Side Inspection," Revision 3
EPS-001-W, "Steam Generator ECT Data Analysis For Waterford 3," Revision 1
QAP-393, "Manual Ultrasonic Examination of Welds in Vessels," Revision 3
NDE 9.04, "Ultrasonic Examination of Ferritic Piping for ASME Section XI," Revision 3
NDE 9.31, "Magnetic Particle Examination (MT) for ASME Section XI," Revision 3
NDE 9.40, "Liquid Penetrant Examination (PT)," Revision 1
NDE 9.41, "Liquid Penetrant Examination (PT) for ASME Section XI," Revision 1

Miscellaneous Documents

ER-W3-2003-0534-000, "Steam Degradation Assessment and Repair Criteria for RF12"

Eddy Current Acquisition Technique Sheets
WTR-01-03
WTR-A-03

**IP 71111.11B**

Procedures:
TQ-201, Systematic Approach to Training Process, Revision 1
TQ-202, Simulator Configuration Control, Revision 1
DG-TQ-201, Design and Development Phase, Revision 2
DG-TQ-201, Implementation Phase, Revision 1
DG-TQ-201, Evaluation Phase, Revision 3
DG-TRNW-001, Operator Training Simulator Deskguide, Revision 8
DG-TRNW-003, Operator Training Examination Development and Administration, Revision 6
TDG-SIM-003, Simulator Steady State and Transient Testing, Revision 1
TDG-SIM-016, Configuration Management, Revision 6
TDG-SIM-017, Conducting a Simulator Outage, Revision 1

Simulator Documents:
Simulator Fidelity Report for 2003
Annual Performance Testing Data for 2002
Transient data
Steady State data
Plant Data from Main Turbine Trip on 14 February 2003
Plant Data from Loss of 2B RCP in 1999
Core Performance Data

Miscellaneous:
Licensed operator annual/biennial examination development model
Licensed operator requal sample plan and two-year guide
Biennial exam testable subject matter
Training Review Group Meeting Minutes, June 4, 2002
Training Review Group Meeting Minutes, September 9, 2002
Training Review Group Meeting Minutes, January 7, 2003
Training Review Group Meeting Minutes, February 25, 2003
Training Review Group Meeting Minutes, June 3, 2003

Simulator Scenarios:
E-68
E-70
E-71
E-91
P-76

Job Performance Measures:
SRO-EP-EMERG-1
RO-CPC-NORM-11
RO-CS-EMERG-7
NAO-SDC-NORM

NAO-CED-OFFNORM-2
RO-PPO-OFFNORM-5

Written Examinations:
WWEX-LOR-03061R
WWEX-LOR-03061S
WWEX-LOR-03062R
WWEX-LOR-03062S

Training Evaluation Reviews:
WLP-OPS-SAF00, 2/28/2002
WLP-OPS-REQ22, 2/25/02
WLP-LOR/AOR-REQ21, 2/26/02
WLP-LOR/PPO30, 5/29/02
WSEM-OPS-COACH, 2/28/02
WLP-LOR-LOG00; 8/22/02
WLP-TYH11; 7/8/02
WLP-LOR-PPO020; 5/29/02
WLP-LOR-TYR09; 5/29/02
WLP-LOR-PPE20; 7/1/02
WLP-OPS-CLR00; 2/28/02
WLP-OPS-SP00; 7/10/02
WLP-OPS-CED00; 5/7/03
WLP-LOR-PPO10, PPO40; 2/13/03
WLP-LOR-TYR08; 1/14/03
WLP-OPS-CLR; 7/29/03
WLP-OPS-TS04; 8/21/03
WLP-OPS-RF00; 8/26/03
WLP-OPS-COL; 7/10/03
WLP-OPS-IC01; 6/24/03

Operations Training Coaching Cards for Functional Recovery Procedure Usage:
30097
30119
30331
30353
30354
30808
30836
31068
31767
31800

Management Observation Cards for Technical Specification Recognition:
37335
37456
37496
37612
37614

**IP 711111.20**

Procedures

Operating Procedure OP-901-521, "Severe Weather and Flooding," Revision 3

Surveillance Procedure OP-903-026, “Emergency Core Cooling System Valve Lineup Verification,” Revision 12

Operating Instruction OI-004-000, “Operation Shift Logs,” Revision 28

Administrative Procedure UNT-007-059, “Foreign Material Exclusion,” Revision 2

Operating Procedure OP-901-521, "Severe Weather and Flooding," Revision 3

Surveillance Procedure STA-001-005, “Leakage Testing of Air and Nitrogen Accumulators for Safety Related Valves,” Revision 6

Surveillance Procedure OP-903-119, “Secondary Auxiliaries Quarterly IST Valve Tests,” Revision 7

Surveillance Procedure OP-903-027, “Inspection of Containment,” Revision 6

Administrative Procedure LI-102, “Corrective Action Process,” Revision 2

Corrective Action Documents

CR 2003-2076, CR 2003-2089, CR 2003-3858, CR 2003-3911, CR 2003-3901, CR 2003-3729, CR 2002-0818, CR 1999-0167, CR 1998-1033, CR 2003-3837, CR 2003-3716, CR 2003-3884, CR 2003-3839, CR 2003-3849, CR 2003-3204, CR 2003-3152, CR 2003-3763, CR 2003-3459, CR 2003-3458, CR 2003-3536, CR 2003-2674, CR 2003-2900, CR 2003-0201,CR 2003-2615, CR 2001-0135, CR 2003-0643, CR2003-2589, CR 2003-3515, CR 2003-3897, CR 2003-3379, CR 2003-3523, CR 2003-3533, CR 2003-3400, CR 2003-3425, CR 2003-3142, CR 2003-3083, CR 2003-3082, CR 2000-1250, CR 2003-3130, CR 2003-3110, CR 2003-2863, CR 2003-3508,

Other

Engineering Calculation EC-M88-024, “Accumulator V, VIII, IX and X Calculation,” Revision 3

Engineering Request ER-W3-97-547-00-01, “Safety Function of Target Rock Solenoid Valves and Pressure Regulating Valves in the SC3 Portion of the NG System,” Revision 1

Design Engineering Procedure NOECP-451, “Conducting Engineering Inspection of Reactor Containment Building Protective Coatings

Program Section CEP-IST-001, “Inservice Testing Plan,’ Revision 2

Engineering Request ER-W3-00-0890, “MSIV Design Basis,” Revision 2

NRC Generic Letter 98-04, “Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-Of -Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment, dated July 14, 1998

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Work Order Package
31122, 50334786, 13532, 13531, 33801, 33381, 50285047, 32863, 19905, 28970

**IP 71121.01**

Condition Reports:
CR-WF3-2002-1806, CR-WF3-2002-1851, CR-WF3-2003-322, CR-WF3-2003-814, CR-WF3-2003-1080, CR-WF3-2003-1290,CR-WF3-2003-1405, CR-WF3-2003-1426, CR-WF3-2003-1521,CR-WF3-2003-1602, CR-WF3-2003-2268, and CR-WF3-2003-2607

Procedures:

| | |
|---|---|
| UNT-001-016 | “Radiation Protection, ”Revision 1 |
| RP-103 | “Access Control,” Revision 2 |
| RP-105 | “Radiation Work Permits,” Revision 4 |
| RP-108 | “ Radiation Protection Posting,” Revision 1 |
| HP-001-107 | “High Radiation Area Access Control”, Revision 16 |

Radiation Work Permits:
2003-1502 RCP 1B Seal Replacement
2003-1613 Replacement of Pressurizer Heaters
2003-1702 Reactor Disassembly

Self-Assessment and Quality Assurance:
W3F3-2003-0012
QS-2003-W3-002
QS-2003-W3-013

**IP 71121.02**

Radiation Work Permits

| | |
|---|---|
| 2003-1511 | Steam Generator Primary Side Work |
| 2003-1512 | Steam Generator Secondary Side Work |
| 2003-1600 | Health Physics Surveys and Job Coverage |
| 2003-1610 | Erect/Dismantle Scaffolding in RCB |
| 2003-1705 | Reactor Re-Assembly |
| 2003-1713 | Work involving Non-Destructive Examination under Reactor Head Shield Frame |

Procedures

| | |
|---|---|
| RP-102 | Radiological Control, Revision 3 |
| RP-105 | Radiation Work Permits, Revision 4 |
| RP-109 | Hot Spot Program, Revision 0 |
| RP-110 | ALARA Program, Revision 1 |
| RP-205 | Prenatal Monitoring, Revision 2 |
| HP-001-101 | ALARA Program Implementation, Revision 13 |
| HP-001-114 | Installation of Temporary Shielding, Revision 8 |

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2002-1616, 2002-1759, 2003-0396, 2003-0535, 2003-1853, 2003-1936, 2003-2211, 2003-2989, 2003-3168, 2003-3253, 2003-3282, 2003-3286, 2003-3361, 2003-3405, 2003-3703, 2003-3718, and ECH-2003-0347

Self-Assessment and Quality Assurance

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|---|---|
| W3F3-2003-0012 | Radiation Protection |
| W3F3-2003-133 | RWP Revisions |
| QS-2002-W3-092 | RWP/ALARA Radiation Practices |
| WT-ECH-2003-074 | RF12 Radiation Protection Outage Readiness |

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**TI 2515/150**

Procedures:
QAP-410, Reactor Vessel Head VT Examination (Alloy 600), Revision 2

MRS-SSP-1534, Reactor Vessel Head Penetration Inspection Tool Operation, Revision 0

WDI-STD-101, RVHI Vent Tube J-Weld Eddy Current Examination, Revision 2

WDI-ET-003, IntraSpect Eddy Current Imaging Procedure for Inspection of Reactor Vessel head Penetrations, Revision 5

WDI-ET-004, IntraSpect Eddy Current Analysis Guidelines for Inspection of Reactor Vessel Head Penetrations, Revision 3

WDI-UT-010, IntraSpect Ultrasonic Procedure for Inspection of Reactor Vessel Head Penetrations, Time of Flight, Longitudinal Wave & Shear Wave, Revision 6

WDI-UT-011, IntraSpect NDE Procedure for Inspection of Reactor Vessel Head Vent Tubes, Revision 4

WDI-UT-013, CRDM.ICI Analysis Guidelines, Revision 3

WDI-STD-122, RVHI CEDM Bottom OD Inspection, Revision 0

WCAL-02, Pulser/Receiver Linerarity Procedure, Revision 2

Calculation:
ECM03-010, Calculation of RPV Head Effective Degradation Years

## LIST OF ACRONYMS

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|---|---|
| NRC | Nuclear Regulatory Commission |
| CFR | Code of Federal Regulations |
| ECT | eddy current testing |
| NRR | Nuclear Reactor Regulation |
| MSIV | main steam isolation valve |