UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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February 11, 2003

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

SUBJECT: WATERFORD 3 - NRC INSPECTION REPORT 50-382/02-05

Dear Mr. Venable:

On December 20, 2002, the NRC completed a team inspection at your Waterford Steam Electric Station, Unit 3. The enclosed report documents the inspection findings, which were discussed on December 20, 2002, and January 31, 2003, with you and other members of your staff.

This inspection examined activities conducted under your license as they relate to the identification and resolution of problems, 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.

Overall, the team found that your process to identify, prioritize, evaluate, and correct problems was generally effective during calender years 2001 and 2002. The team reviewed 250 condition reports that were opened or closed during the period and found, in general, that station personnel effectively identified, characterized, and prioritized problems. A number of issues were identified associated with the evaluation and correction of degraded conditions by the team. Most of these issues were identified when the team reviewed corrective actions associated with longstanding degraded conditions at your facility and had cross-cutting aspects in the area of problem identification and resolution.

Based on the results of this inspection, the NRC has identified issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations are associated with these issues. These violations are being treated as noncited violations, consistent with Section VI.A of the Enforcement Policy. These noncited violations are described in the subject inspection report. If you contest the violation or significance of these noncited violations, 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, 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).

If you have any questions about these issues discussed in this report, please contact me at (817)860-8159.

Sincerely,

/**RA**/

Anthony T. Gody, Chief
Operations Branch
Division of Reactor Safety

Docket: 50-382
License: NPF-38

Enclosure:
NRC Inspection Report
50-382/02-04

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-382
License:	NPF-38
Report:	50-382/02-04
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	September 29 through December 28, 2002
Team Leader:	M.C. Hay, Senior Resident Inspector, Project Branch E
Inspectors:	L. Ricketson, Senior Health Physicist, Plant Support Branch H. Bundy, Senior Operations Engineer, Operations Branch T. Jackson, Resident Inspector, Project Branch E
Approved By:	Anthony T. Gody, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR05000382-02-05; Entergy Operations, Inc.; on 12/09/02-12/20/02; Waterford Steam Electric Station; Unit 3; Identification and Resolution of Problems, Mitigating Systems, Barrier Integrity

The inspection was conducted by a senior resident inspector, a senior operations engineer, a senior health physicist, and a resident inspector. The significance of most findings is indicated by their color (green, white, yellow, red) using IMC 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be “green” or 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. Inspector Identified and Self-Revealing Findings

Identification and Resolution of Problems

The licensee's process to identify, prioritize, evaluate, and correct problems was generally effective during calender years 2001 and 2002. The team reviewed 250 condition reports that were opened or closed during the period and found, in general, that station personnel effectively identified, characterized, and prioritized problems. Some issues involving the evaluation and correction of degraded conditions were identified by the team. Most of these issues were associated with longstanding degraded conditions that were identified and corrected by the licensee during this period and included the following: (1) an untimely identification of a void condition in the containment spray system existing between April and September 2002, (2) inadequate extent of condition reviews to identify main steam flow venturi degradation which existed since 1995 and the deleterious affect an oil coating which existed since 1997 would have on electrical components associated with the emergency diesel generator, (3) the inappropriate use of engineering analyses that allowed piping supports to exceed design basis allowable stresses during postulated accidents with voids in the low pressure safety injection system since 1997, (4) an inadequate verification of the design adequacy of a plant modification to vent low pressure safety injection system voids installed in June 2002, and (5) untimely corrective actions which resulted in a forced shutdown to repair weld cracks in the charging system in March 2000. Most of these issues had cross-cutting aspects in the area of problem identification and resolution.

Cornerstone: Mitigating Systems/Barrier Integrity

Green. Three examples associated with failures to adequately evaluate the extent of conditions adverse to quality were identified. The failure to promptly identify and correct these degraded conditions was a violation of 10 CFR Part 50, Appendix B, Criterion XVI (Section 4OA2.b). Three examples included:

- The licensee failed to promptly identify and correct a degraded condition resulting in the electrical and electronic components inside Emergency Diesel Generator B control cabinet being subjected to oil intrusion since 1997. The team found that the licensee failed to evaluate the cause of the oil intrusion until 2001, took no corrective actions in 2001 or 2002 to prevent the oil intrusion when the source was identified, and failed to fully evaluate the detrimental effects that the oil intrusion could pose to the electrical and electronic components.

The failure to promptly identify and correct the degraded condition resulting in the electrical and electronic components inside Emergency Diesel Generator B control cabinet being subjected to oil intrusion since 1997 was determined to be a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This finding is greater than minor because if left uncorrected it would become a more significant safety concern. This finding is of very low safety significance since the degraded condition did not result in a loss of the emergency diesel generator safety function.

- The licensee failed to promptly identify and correct a degraded condition resulting in exceeding the rated thermal power limit from February 1995 to March 2002. This condition, identified by the licensee in March 2002, introduced non-conservative excore neutron detector calibration errors which affected the high linear power level, high logarithmic power level, high local power density, and low departure from nucleate boiling ratio, reactor protection trip functions.

  The failure to promptly identify and correct the overpower condition was determined to be a violation of the facility operating License NPF-38 and 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This finding is greater than minor because it affected four reactor trip functions in a non-conservative manner, thus, potentially impacting the barrier cornerstone integrity. The finding is of very low safety significance since it was determined that the accident analysis, Chapter 15 of the Final Safety Analysis Report, bounded the non-conservative trip functions. This finding is also of very low safety significance since actual fuel barrier integrity was never challenged during the overpower condition.

- On April 18, 2002 when the low pressure safety injection Train B was found voided, the licensee failed to identify that the containment spray system Train B would also be voided from similar plant conditions. The containment spray voiding was identified by the licensee on September 17, 2002, when abnormal indications were noted by operators during a surveillance. Action was then taken by the licensee to correct the degraded condition. However, the licensee failed to identify the degraded condition during previous opportunities.

  The failure to promptly identify and correct the voided condition affecting containment spray Train B was determined to be a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This finding is greater than minor because if left uncorrected the voided condition could impact the reliability of the containment spray system to perform its safety function during accident conditions. The finding is of very low safety significance since the licensee could demonstrate through analysis that the actual degraded condition found would not have prevented the system from performing its safety function during accident conditions.

Cornerstone: Mitigating Systems

Green. Two examples of failures to implement timely corrective actions to resolve degraded conditions were identified. The failure to promptly identify and correct these degraded conditions was a violation of 10 CFR Part 50, Appendix B, Criterion XVI (Section 4OA2.c). Two examples included:

- The licensee failed to promptly identify and correct piping connections susceptible to fatigue stress cracking resulting in an unisolable leak from the charging system header on March 6, 2000. In 1997, the licensee experienced a crack of the charging system header due to fatigue stress cracking and determined additional piping connections were susceptible. The piping connection that failed in March 2000 was identified as being susceptible to fatigue stress cracking, however, no corrective actions had been taken.

  The failure to promptly identify and correct piping susceptible to fatigue stress cracking resulting in an unisolable leak from the charging system header on March 6, 2000, is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. The finding is greater than minor because if left uncorrected the finding could become a more significant event. The finding is of very low safety significance since the degradation of the system was identified and corrected prior to the safety function of the system being adversely impacted.

- The licensee failed to promptly implement timely corrective actions to operate and maintain the low pressure safety injection system as described in the Final Safety Analysis Report. Specifically, since 1997, the licensee utilized multiple analysis for evaluating degraded piping and pipe supports to evaluate acceptable void sizes. These analysis utilized allowable stresses that exceeded the design criteria allowable stresses described in the facilities Final Safety Analysis Report for the low pressure safety injection system.

  The failure to implement timely corrective actions to restore and maintain the low pressure safety injection system as described in the Final Safety Analysis Report is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. The finding is greater than minor because the Mitigating Systems Objective to ensure the availability, reliability, and capability is potentially affected when the system is maintained outside of its design criteria as described in the

Final Safety Analysis Report. The finding is of very low safety significance since the analysis used to assess the degraded condition ensured the system could perform its safety function.

Cornerstone: Barrier Integrity

- Green. The licensee failed to maintain design control of the low pressure safety injection system, Train A, in accordance with the design basis, as described in the Final Safety Analysis Report, when installing a modification to mitigate adverse voiding conditions that have affected the system. The failure to maintain design control of the system resulted in loss of a Seismic Class 1, ASME Section III, Safety Class 2, barrier during post accident conditions.

  The failure to maintain design control of the system is a violation of 10 CFR Part 50, Appendix B, Criterion III. This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This issue screens more than minor because the Barrier Integrity Objective to provide reasonable assurance that the physical design barriers protect the public from radionuclide releases caused by accidents or events was potentially affected. The finding is of very low safety significance since only degradation of the radiological barrier function provided for the auxiliary building was affected.

Report Details

4 OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

a. Effectiveness of Problem Identification

(1) Inspection Scope

The inspectors reviewed items selected across the seven cornerstones to determine if problems were being properly identified, characterized, and entered into the corrective action program for evaluation and resolution. Specifically, the team's review included a selection of approximately 250 condition reports that were opened or closed from May 2001 through November 2002. The team also reviewed a sample of licensee audits and self assessments, trending reports, system health reports, and various other reports and documents related to the problem identification and resolution program. The audit and self-assessment results were compared with the self-revealing and NRC-identified issues to determine the effectiveness of the audits and self assessments.

The team interviewed station personnel and evaluated corrective action documentation to determine the licensee's threshold for identifying problems and entering them into the corrective action program. The team evaluated the licensee's efforts in establishing the scope of problems by reviewing operational logs, action plans, maintenance action items, and results from surveillance tests.

In addition, the team reviewed the licensee's evaluation of selected industry experience information, including operator event reports and NRC and generic notices, to assess if issues applicable to the Waterford Steam Electric Station, Unit 3, were appropriately addressed.

(2) Issues

The inspectors determined that problems were effectively identified, characterized, and entered into the licensee's corrective action program. During interviews, plant personnel indicated that they believed that a low threshold for entering problems into the corrective action program had been established. Licensee audits and self assessments were appropriately comprehensive.

b. Prioritization and Evaluation of Issues

(1) Inspection Scope

The team reviewed approximately 250 condition reports and supporting documentation, including root cause evaluations, to ascertain whether the licensee identified and considered the full extent of conditions, generic implications, common causes, and previous occurrences. In addition, the inspectors reviewed licensee evaluations of selected industry operating experience information, including operating event reports and NRC and vendor generic notices, to assess if issues applicable to the Waterford Steam Electric Station, Unit 3, were appropriately addressed.

(2) Issues

The previous NRC Identification and Resolution of Problems team inspection noted that human performance was a significant contributor to conditions adverse to quality reviewed between June 1, 2000, and May 31, 2001. The team noted that the licensee's corrective actions to address this concern included a program to monitor both departmental and site-wide human performance errors and precursors to errors. The team independently reviewed condition reports related to human-performance errors and concluded that the licensee's actions appeared appropriate.

Some evaluations for conditions entered into the corrective action process failed to fully evaluate the extent of condition resulting in ineffective corrective actions. These examples and others contained in Section c. below are indicative of a cross-cutting issue in problem identification and resolution.

Example 1

Prior to the licensee's last refueling outage (refueling outage 11) in March 2002, voided conditions were discovered on multiple occasions in the suction header of both shutdown cooling trains. These voids were evaluated by the licensee in Condition Reports 2001-1348 and 2002-0052 initiated on December 13, 2001, and January 10, 2002, respectively. The voids were found to exist at Primary Containment Penetrations 40 and 41, each displacing approximately 250 gallons of water. Corrective actions included filling and venting the system and declaring the system operable. The evaluation of these conditions adverse to quality failed to recognize that filling the voided sections of piping could result in thermal binding concerns associated with Safety Injection Valves SI-405(A) and SI-405(B). Subsequently, on March 23, 2002, the valves failed to open during efforts to establish shutdown cooling resulting in declaration of an Alert. This issue was dispositioned as a noncited violation in NRC Inspection Report 50-382/02-03. The evaluations also failed to recognize that the voided conditions found at Penetrations 40 and 41 resulted in a potential for exceeding the total allowable primary containment leakage. The inboard and outboard primary containment isolation valves that isolate Penetrations 40 and 41 were exempt from Appendix J leak rate testing. This exemption was based on the licensee's position that the penetrations would remain full of water during post-accident conditions and that a monitoring program would be established to ensure that a water seal was being maintained. The licensee subsequently recognized this condition adverse to quality during Refueling Outage 11 and performed leak-rate tests that demonstrated the total allowable leakage outside containment was not exceeded. The team noted that the licensee subsequently included the subject primary containment isolation valves into their Appendix J leak-rate testing program until corrective actions have been implemented that will maintain the required water seal.

Example 2

Introduction: The licensee failed to promptly identify and correct a degraded condition resulting in the electrical and electronic components inside Emergency Diesel Generator B control cabinet being subjected to oil intrusion since 1997. The licensee failed to evaluate the cause of the oil intrusion until 2001, took no corrective actions in 2001 to prevent the oil intrusion when the source was identified, and failed to fully evaluate the detrimental effects that the oil intrusion could pose to the electrical and electronic components. The issue was determined to be a violation of 10 CFR Part 50, Appendix B, Criterion XVI and was characterized under the significance determination process as having very low safety significance.

Description: In April 1997, during Refueling Outage 8, the licensee discovered a film of oil that covered various electrical and electronic components inside the generator control cabinet for Emergency Diesel Generator B. The licensee documented the condition in Condition Report 1997-1047. The source of the oil was unknown and it covered electrical and electronic components, such as the ventilation filter and fan, electrical terminal strips, wires, silicon-controlled rectifiers, relays, and other components that are used to control generator field voltage. The operability analysis in Condition Report 1997-1047 stated that the oil would not prevent the diesel generator from accomplishing its required function and the oil did not pose a fire hazard because it would not ignite in the presence of a spark. The evaluation also stated that since the diesel passed it surveillance tests, there was no impact on electrical equipment operation due to oil. However, the team noted that the licensee did not address the cooling capability of the oil-covered components, the corrosive potential of the oil on the electrical/electronic components, and other long-term impacts the oil may have on the electrical/electronic components not necessarily rated for a petroleum environment.

In May 2001, during a preventive maintenance activity on Emergency Diesel Generator B, Condition Report 2001-0595 was initiated documenting an oil film on electrical/electronic components inside the generator control cabinet. The affected components and the amount of oil appeared to be similar to that found in 1997. Condition Report 2001-0595 noted that in Refueling Outage 9, a light film of oil was found on the same electrical/electronic components, but it was not documented in a condition report. The licensee determined the oil was Mobile Delvac 1340, which was used in the emergency diesel generator crankcase and starting air compressors. It was later determined that the source of the oil was a leak through the diesel end-housing shaft felt seal. The oil leaked onto the flywheel, and when the diesel ran, the oil would be slung into the ventilation inlet of the generator control cabinet. The oil would collect on the corrugated metal dust filter, pass through the filter, and be slung by the cabinet ventilation fan onto electrical/electronic components inside the cabinet.

The corrective actions detailed in Condition Report 2001-595 included a quarterly replacement of the corrugated metal dust filter and continued monitoring of the oil accumulation. Repair of the shaft felt seal leakage was determined to be difficult due to constraints on rigging and realignment of the diesel shaft during repairs. The team noted that the corrective actions associated with Condition Report 2001-595 were not effective at preventing oil intrusion of the emergency diesel generator electrical components.

The team determined that ineffective corrective actions since 1997 allowed a degraded condition to exist that could impact the long-term operability of Emergency Diesel Generator B. In particular, the licensee identified an oil film on electrical/electronic components in Emergency Diesel Generator B generator control cabinet during Refueling Outage 8. The licensee failed to adequately evaluate the impact of the oil film on the components with respect to cooling, interference with component operation, and corrosion. Additionally, the licensee failed to identify the source of the oil intrusion from 1997 through 2001. Corrective actions associated with Condition Report 1997-1047 failed to prevent the oil from continuing to be applied to the components as verified by the observation made in Condition Report 2001-595. Additionally, the licensee failed to document the oil film found during Refueling Outage 9, even though a part of the corrective actions was to monitor the oil film during refueling outages.

After the team's question, the licensee initiated Condition Report-WF3-2002-2043 to fully evaluate the operability impact of the oil film on components inside Emergency Diesel Generator B generator control cabinet. The team concluded that past and current functionality of Emergency Diesel Generator B were not impacted by the oil film. However, as documented in Condition Report-WF3-2002-2043 and discussed with engineers, there is a potential long-term impact of the oil film with respect to corrosion of wire insulation, reduced life span of certain components due to reduced cooling, and the potential for oil to wick into components and prevent proper operation.

<u>Analysis</u>: The team determined that this finding was more than minor since if left uncorrected it would become a more significant safety concern. Specifically, the licensee failed to evaluate the full extent of the condition and initiate appropriate corrective actions from 1997 to 2001 that would prevent an oil film from forming on Emergency Diesel Generator B generator control cabinet components and impact the long-term functionality of the emergency diesel generator. The team determined that past and current functionality of Emergency Diesel Generator B was not impacted by the oil film and that surveillance tests, including a 24-hour run, verified that proper operation of Emergency Diesel Generator B. Using the Significance Determination Process, as described in Inspection Manual Chapter 0609, under the Mitigating System Column, the finding screens as Green since the deficiency was confirmed not to result in a loss-of-safety function.

<u>Enforcement</u>: 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," states, in part, that "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Contrary to the above, the licensee failed to identify the impact of and correct an oil film from forming on Emergency Diesel Generator B generator control cabinet components that could impact the long-term operation of Emergency Diesel Generator B. The team determined this to be the first example of a Criterion XVI violation resulting from the failure to fully evaluate the extent of a degraded condition. This violation is being treated as a noncited violation (50-382/0205-01) consistent with Section VI.A of the NRC Enforcement Policy. The licensee documented this issue in their corrective action process as Condition Report 2002-2043. The licensee indicated that corrective actions to prevent recurrence include a future modification to preclude leaking oil from entering the electrical cabinets until the leaking seal can be repaired.

Example 3

Introduction: The licensee failed to promptly identify and correct a degraded condition resulting in exceeding the rated thermal power limit from February 1995 to March 2002. This condition introduced non-conservative excore neutron detector calibration errors affecting the high linear power level, high logarithmic power level, high local power density, and low departure from nucleate boiling ratio, reactor protection trip functions. The issue was determined to be a violation of 10 CFR Part 50, Appendix B, Criterion XVI and was characterized under the significance determination process as having very low safety significance.

Description: The Waterford Steam Electric Station, Unit 3, uses the indicated reactor power level from the core operation limit supervisory system (COLSS) to ensure plant operation thermal limits are protected and to calibrate the excore neutron detectors. To perform this function, COLSS calculates actual plant power by using the mass flow and enthalpy of main steam, steam generator blowdown, feedwater, and the credits and losses associated with the primary reactor coolant system. Prior to February 1995, actual plant power was calculated using the feedwater venturis to measure feedwater mass flow (FWBSCAL). The licensee suspected that the feedwater venturis were fouled, which caused COLSS to indicate a higher reactor power level than actual. To address the suspected feedwater venturi fouling, the licensee modified COLSS to calculate actual power using the main steam venturis (MSBSCAL). As part of the modification, the licensee calculated new mass flow constants that were derived from a strap-on ultrasonic feedwater flow meter. After the COLSS modification, power calculated using MSBSCAL was 0.7 to 0.8 percent higher than the power calculated using FWBSCAL.

In August 1996, the licensee received numerous “COLSS MSBSCAL VALIDITY CHECK” alarms from COLSS, and Condition Report-WF3-1996-1299 was initiated to investigate the cause of the alarms. Upon calibration of the steam and feedwater flow transmitters, the licensee concluded that the most probable root cause for the alarms was additional fouling of the feedwater venturis. The COLSS MSBSCAL VALIDITY CHECK alarm limit was revised to allow a 2.5 percent power difference between MSBSCAL and FWBSCAL. Prior to the revision of the alarm limit, a deviation of 1.7 percent was allowed between MSBSCAL and FWBSCAL. The licensee attributed the cause of the alarms to additional feedwater venturi fouling despite the fact that there was a net megawatt electric generation and first stage turbine pressure increase that indicated an actual reactor power increase. Additionally, the licensee did not sufficiently review the secondary calorimetric calculation based on main steam venturi data to assure its validity.

In April 2002, during Refueling Outage 11, a leading edge flow meter was installed into both feedwater lines, with the intention of using this instrument to indicate actual plant power. The leading edge flow meter is reported to have an accuracy error of approximately less than 0.27 percent and was used to support a one percent power uprate for the Waterford Steam Electric Station, Unit 3. The licensee anticipated the COLSS reactor power indication using the leading edge flow meter (USBSCAL) would closely match power calculated by MSBSCAL. However, upon activation of USBSCAL, it deviated from MSBSCAL by 1.9 percent versus a deviation from FWBSCAL by only

0.3 percent. Therefore, the licensee concluded that from February 1995 to March 2002, the time when MSBSCAL was used to calculate actual power, that the licensed power limit of 3390 megawatt thermal was exceeded by 0.7 to 1.9 percent. The root-cause analysis contained in Condition Report-WF3-2002-0824 lists the most probable cause as main steam venturi erosion/corrosion and two contributing causes. Contributing causes include incorrect MSBSCAL mass-flow constants that were used during the 1995 COLSS modification. The incorrect mass-flow constants resulted in an approximate 1.2 percent non-conservative power bias in MSBSCAL. Another contributing cause was the failure to identify the overpower condition in the root-cause analysis contained in Condition Report-WF3-1996-1299.

The team determined that the margin in the accident analysis, as described in Chapter 15 of the Final Safety Analysis Report, and the tendency for the excore neutron detectors to drift in the conservative direction enveloped the degraded condition associated with calibrating the excore neutron detectors with a non-conservative reactor power indication.

The team determined that the licensee failed to effectively and critically evaluate indications that they had exceeded their licensed power limit, such that the condition existed for approximately 7 years. In August 1996, the licensee conducted a root cause analysis into the cause of COLSS MSBSCAL VALIDITY CHECK alarm, but they did not thoroughly consider the potential for MSBSCAL calculations to be incorrect, neither did they assess indications that reactor power had actually increased. One impact of the licensee's failure to identify the overpower condition is the calibration of excore neutron detectors with the non-conservative reactor power indication. This impact resulted in a degraded condition associated with the reactor protection system for approximately the same time the overpower condition existed. The affect of the overpower condition on excore neutron detectors was not discussed in Condition Report-WF3-2002-0824 or other documents in the licensee's corrective action system. Upon identifying the overpower condition in March 2002, the licensee corrected the condition, including a calibration of excore neutron detectors to the appropriate power level.

<u>Analysis</u>: Using the guidance in Appendix B of Inspection Manual Chapter 0612, this issue impacts the Barrier Integrity Cornerstone, and in particular, the Fuel Clad Design Control Objective. The finding screens as more than minor since calibrating the excore neutron detectors with the non-conservative reactor power indication would cause the affected reactor protection functions to trip at a non-conservative reactor power level. Using the Significance Determination Process Phase 1 screening worksheet, found in Appendix A of Inspection Manual Chapter 0609, the issue screens as Green for the following reasons: (1) during the years where the non-conservative trip setpoints were applied, the condition was bounded by the accident analysis, (2) the excore neutron detector readings tend to shift towards the conservative direction throughout core life due to tendency for more neutrons to reach the excore detectors as core burn-up occurs, and (3) actual fuel barrier integrity was never challenged during the overpower condition.

Enforcement: Waterford Steam Electric Station, Unit 3, Facility Operating License No. NPF-38, Section 2.C.1, stated, in part, that the facility is authorized to operate at reactor core power levels not in excess of 3390 megawatts thermal. Criterion XVI, Appendix B of 10 CFR Part 50, "Corrective Actions," states, in part, that "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected." The failure to promptly identify and correct the overpower condition resulting in adversely affecting several reactor protection trip functions from February 1995 to March 2002 is a violation of the facility operating License NPF-38 and 10 CFR Part 50, appendix B, Criterion XVI. The team determined this to be the second example of Criterion XVI violation resulting from the failure to fully evaluate the extent of a degraded condition. This violation is being treated as a noncited violation (NCV 50-382/0205-01) consistent with Section VI.A of the NRC Enforcement Policy. The licensee documented this issue in their corrective action process as Condition Report 2002-0824.

Example 4

Introduction: The licensee failed to promptly identify and correct a degraded condition resulting in a voided condition of Containment Spray Train B from April 18, 2002, through September 17, 2002. The licensee failed to identify the voided condition in April 2002 when the Low Pressure Safety Injection Train B was found voided for the same root cause. Additionally, operators failed to identify the voided condition on three separate occasions when abnormal indications were not recognized during surveillance activities. The issue was determined to be a violation of 10 CFR Part 50, Appendix B, Criterion XVI and was characterized under the significance determination process as having very low safety significance.

Description: On September 17, 2002, operations personnel noted that there was an abnormal drop in level indication on Containment Spray Train A riser when stroking containment spray valve CS-125A during performance of OP-903-121, “Safety Systems Quarterly IST Valve Tests.” Condition Report 2002-1539 was initiated. Engineering personnel determined that a void was the most probable cause of the abnormal indications. Ultrasonic testing confirmed that a void was present. The void size was determined to be approximately 3 cubic feet and was located in horizontal runs of the system between the pump and the riser header. The team noted that the system was designed to remain full of water.

The licensee’s investigation of this event resulted in the discovery that on multiple occasions prior to September 17, 2002, the abnormal drop in level indication was seen by operating personnel. These abnormal indications were not recognized by operators on April 19, May 5, and June 25, 2002. The team noted that a contributing factor to operators not understanding the abnormal indication was due to procedural guidance that indicated slight changes in riser level were expected during the surveillance. The root-cause analysis determined that voiding resulted from inadequate venting of the containment spray system following Refueling Outage 11, that ended on April 16, 2002. The team noted that during shutdown conditions a portion of the containment spray system and low pressure safety injection system comprise the shutdown cooling system, and that there was a documented history of the licensee observing the introduction of gasses into solution during shutdown cooling operations.

The team noted that following Refueling Outage 11, on April 18, 2002, Low Pressure Safety Injection Train B was found to contain several voids that totaled approximately 3.3 cubic feet. This condition was documented in Condition Report 2002-00818. The team noted that the root-cause analysis for this voiding event determined that an inadequate vent plan was in place to remove gasses from the low pressure safety injection system once the system was realigned from the shutdown cooling to the safety injection mode. The team reviewed the corrective actions for this event and noted that the licensee had failed to recognize that the containment spray system could potentially have voiding since a portion of this system is also used in the shutdown cooling line up. The failure to recognize that the containment spray system was also susceptible to same voiding mechanism that affected the low pressure safety injection system resulted in the failure to promptly identify the voids in Containment Spray Train B.

Analysis: Using the guidance in Appendix B of Inspection Manual Chapter 0612, this issue screens more than minor since if left uncorrected the voided condition had the potential to impact the reliability of the containment spray system to perform its safety function during accident conditions. Using the Significance Determination Process Phase 1 screening worksheet, found in Appendix A of Inspection Manual Chapter 0609, the issue screens as Green since the licensee could demonstrate through analysis that the degraded condition found would not have prevented the system from performing its safety function during accident conditions.

Enforcement: 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” states, in part, that “Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected.” The failure to promptly identify and correct the voiding condition affecting Containment Spray Train B, from April 18 through September 17, 2002, is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. The team determined this to be the third example of a Criterion XVI violation resulting from the failure to fully evaluate the extent of a degraded condition. This violation is being treated as a noncited violation (NCV 50-382/0205-01) consistent with Section VI.A of the NRC Enforcement Policy. The licensee documented this issue in their corrective action process as Condition Report 2002-1539.

Example 5

The team noted that Condition Report 2002-818 documented voided conditions affecting Low Pressure Safety Injection Train B that exceeded the acceptable void size that was calculated using Generic Letter 91-18 guidance. Generic Letter 91-18 guidance allows the evaluation of degraded piping and pipe supports using allowable stresses that exceed the systems original design allowable stresses. The reportability section of the condition report stated that the event was not reportable because the system was inoperable less than the allowed outage time of 7 days. The team noted that since the void size exceeded the degraded, but operable analysis, then low pressure safety injection could potentially have been susceptible to structural damage if called upon to mitigate the consequences of an accident. The team questioned the licensee about whether this could pose an unanalyzed condition that could significantly degrade plant safety. Upon questioning, the licensee indicated that no analysis had been performed to address this concern. Subsequently, engineering reviewed the degraded condition and adequately demonstrated, based on actual plant conditions during the time the void was present, that structural damage would not result. Although this example is used as an example of an inadequate evaluation, no violation of requirements existed because the licensee's subsequent analysis concluded no structural damage would result.

Example 6

The team reviewed 18 licensee evaluations of selected industry experience information. The team noted that the licensee had appropriately evaluated the applicability of the information with respect to Waterford Steam Electric Station, Unit 3. However, the team did note two examples were the licensee's evaluations appeared narrowly focused. Both of these examples were discussed with the licensee and received additional evaluation.

The first example involved Information Notice 2001-06, "Centrifugal Charging Pump Thrust Bearing Damage Not Detected Due to Inadequate Assessment of Oil Analysis Results and Selection of Pump Surveillance Points," issued May 11, 2001. Information Notice 2001-06 emphasized the need to pursue abnormalities observed in inservice and technical specification surveillance tests even though the test runs may meet the current acceptance criteria. The example in the information notice illustrated how the oil analysis for a centrifugal charging pump indicated a high particulate concentration, although the pump met all the acceptance criteria for the oil analysis, inservice tests, and surveillance tests. Subsequently, a significantly damaged thrust bearing was identified when a mechanical seal was replaced on a later date. The damaged thrust bearing was not captured by the testing program because abnormalities in the test were not investigated and selection of test points would not reveal the thrust bearing problem.

The licensee performed an impact review of Information Notice 2001-06 and documented it in LO-OPX-2001-071. The licensee considered the positive-displacement charging pumps, the high-pressure injection pumps, and the low-pressure injection pumps all applicable to the issues described in the Information Notice. However, the licensee did not consider other pumps, such as emergency feedwater pumps, component cooling water pumps, containment spray pumps, and auxiliary component cooling water pumps because they ran at a lower speed than the centrifugal pump example provided in the information notice. The team noted that the Information Notice was not limited to high-speed pumps only and that the generic implications could be extended to other components such as diesel generators and valves, which also require inservice and/or surveillance tests.

The second example involved the licensee's evaluation of Information Notice 2002-03, "Highly Radioactive Particle Control Problems During Spent Fuel Pool Cleanout," issued January 10, 2002. Information Notice 2002-03 emphasized that radioactive particles can present not only shallow-dose risks, but at higher activity levels, whole body dose risks. The information notice also pointed out that a licensee had underestimated the contact dose rates of discrete radioactive particles when using conventional hand-held survey instruments.

The licensee did not conduct an impact evaluation to determine the applicability of the information in Information Notice 2002-03. Instead, the licensee took credit for the impact evaluation performed earlier, in response to Significance Event Report 3-01, issued March 30, 2001. This action did not take into consideration the fact that, while both the Information Notice 2002-03 and Significant Event Report 3-01 discussed the same event, the lessons-learned presented by the two documents were not identical. Further, the impact evaluation conducted in response to Significant Event Report 3-01 was narrowly focused, concluding that the information about discrete radioactive particle controls had no applicability because the licensee would not be cutting boiling water reactor control blades, as in the specific example discussed.

c. Effectiveness of Corrective Actions

(1) Inspection Scope

The team reviewed the condition reports, audits, assessments, and trending reports described in Section 4OA2.a above to verify that corrective actions related to the issues were identified and implemented in a timely manner commensurate with safety, including corrective actions to address common cause or generic concerns. A listing of specific documents reviewed during the inspection is included in the attachment to this report.

The team evaluated the timeliness and adequacy of operability determinations and evaluations. The team reviewed corrective actions planned and implemented by the licensee and sampled specific technical issues to determine whether adequate decisions related to structure, system, and component operability were made.

(2) Issues

The team determined that the majority of conditions adverse to quality were effectively resolved in a timely manner. This conclusion was supported by the relatively few examples of repetitive issues identified, and the declining trend observed for those systems in the maintenance rule category defined in 10 CFR 50.65(a)(1). However, the team did note that several long-standing degraded conditions were not effectively resolved in a timely manner. These conditions included the diesel generator oil leak affecting its electrical components since 1997 as previously discussed in Section b. above, and the following additional examples.

Example 1

Introduction: The licensee failed to promptly identify and correct piping connections susceptible to fatigue stress cracking resulting in an unisolable leak from the charging system header on March 6, 2000. In 1997, the licensee experienced a crack of the charging system header due to fatigue stress cracking and determined additional piping connections were susceptible. The piping connection that failed in March of 2000 was identified as being susceptible to fatigue stress cracking, however, no corrective actions had been taken. The issue was determined to be a violation of 10 CFR Part 50, Appendix B, Criterion XVI, and was characterized under the significance determination process as having very low safety significance.

Description: On March 6, 2000, boric acid buildup in a socket weld adjoining a hydrostatic test connection to Charging Pump A suction piping was documented in Condition Report 2000-0199. The buildup of boric acid was determined to be caused by a 1/4-inch long full penetration crack in the weld. A leak of one to two drops per minute was emanating from the crack. The charging portion of the chemical volume control system was declared inoperable and the plant was shut down per the requirements of Technical Specification 3.0.3.

On September 18, 1997, a similar crack had been discovered on Charging Pump A discharge relief valve vent, Valve CVC-1922A. Design engineering evaluated the piping on all three charging pumps and determined that additional piping connections were susceptible to fatigue stress cracking and recommended a modification to shorten the pipes. The cantilevered hydrostatic test connection on the suction piping for charging Pump A that failed on March 6, 2000, was specifically identified as needing modification. On December 2, 1998, Modification Package ER-98-0946 was approved to remove hydrostatic test connections. Implementation of Modification Package ER-98-0946 was recognized as a Generic Letter 91-18 issue, but was not adequately flagged and tracked to ensure installation at the first opportunity.

Subsequently, the licensee failed to perform the corrective action on the subject weld during the next available opportunity (Refueling Outage 9). The licensee rescheduled the modification for the next refueling outage, however, the failure on March 6, 2000, occurred prior to the outage. The licensee’s root-cause analysis for the issue determined that corrective actions in response to known problems were not completed in a timely manner due to the failure to recognize the urgency of the problem.

Analysis: Using the guidance in Appendix B of Inspection Manual Chapter 0612, this issue screens more than minor since if left uncorrected the finding could become a more significant event. Using the Significance Determination Process Phase 1 screening worksheet, found in Appendix A of Inspection Manual Chapter 0609, the issue screens as green since the degradation of the system was identified and corrected prior to the safety function of the system being adversely impacted.

Enforcement: 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” states, in part, that “Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.” The failure to promptly identify and correct piping susceptible to fatigue stress cracking resulting in an unisolable leak from the charging system header on March 6, 2000, is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. The team determined this to be the first example of a Criterion XVI violation resulting from the failure to implement timely corrective actions to resolve a degraded condition. This violation is being treated as a noncited violation (NCV 50-382/0205-02) consistent with Section VI.A of the NRC Enforcement Policy. The licensee documented this issue in their corrective action process as an update to the original Condition Report 2000-0199.

Example 2

Introduction: The licensee failed to promptly implement timely corrective actions to operate and maintain the low pressure safety injection system as described in the Final Safety Analysis Report. Specifically, since 1997 the licensee utilized multiple analysis for evaluating degraded piping and pipe supports to evaluate acceptable void sizes. These analysis utilized allowable stresses that exceeded the design criteria allowable stresses described in the facilities Final Safety Analysis Report for the low pressure safety injection system. The failure to implement timely corrective actions to restore and maintain the low pressure safety injection system as described in the Final Safety Analysis Report is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This issue was characterized under the significance determination process as having very low safety significance.

Description: The team reviewed Condition Report 2002-00818 pertaining to Low Pressure Safety Injection Train B containing several voids that totaled approximately 3.3 cubic feet. This condition report documented that extensive voiding issues affecting operability of both trains of low pressure safety injection have occurred since 1993. The inspectors noted that voiding conditions have resulted in documented water hammer events, air binding of Low Pressure Safety Injection Pump A, failure of shutdown cooling system valves to open due to thermal locking, loss of required water seals in primary containment penetrations, and numerous voiding conditions that have resulted in declaring the systems inoperable to perform vent and fill evolutions.

The team noted that since December of 2001 the NRC had documented three noncited violations associated with voiding conditions of the low pressure safety injection system. Two of these violations involved inadequate corrective actions to prevent nitrogen saturated water from Safety Injection Tank 2B from leaking into the low pressure safety injection system. The team reviewed the licensee's corrective actions for these violations and noted that extensive effort was being taken to reduce intrusion of nitrogen into the low pressure safety injection system. These actions have included replacement of Safety Injection Valve SI-142A during their previous refueling outage, and installation of automatic vents at Valves SI-133A and SI-1402A. The team also noted modifications to a pipe support and pipe support plate in the Low Pressure Safety Injection A system were being implemented. The team noted that these pipe support modifications were being installed to prevent voids in this system from requiring an evaluation using Generic Letter 91-18 guidance for evaluating degraded pipe and pipe supports when voided conditions were found. The team noted that in 1997 Analysis EC-P97, "Operability Evaluation-LPSI B," was performed that calculated an acceptable void size for Low Pressure Safety Injection Train B. This analysis utilized the guidance contained in Generic Letter 91-18 pertaining to evaluation of degraded pipe and pipe supports allowing the use of ASME Section III, Appendix F, criteria. Appendix F allows for evaluating degraded pipe and pipe supports using stresses that exceed the licensing basis allowable stresses for the system. The team reviewed the guidance in Generic Letter 91-18, which states, in part, "Upon discovery of a nonconformance with piping and pipe supports, licensee's may use the criteria in Appendix F of Section III of the ASME Code for operability determinations. These criteria and use of Appendix F are valid until the next refueling outage when the supports are to be restored to the FSAR criteria." The guidance also states that the NRC expects time frames longer than the next refueling outage to be explicitly justified by the licensee as part of the deficiency tracking documentation. If the licensee does not resolve the degraded or nonconforming condition at the first available opportunity or does not appropriately justify a longer completion schedule, the staff would conclude that corrective action was not timely and would consider taking enforcement action.

The team noted that voiding concerns affecting both trains of low pressure safety injection persisted through 2002. In 2002, another voiding analysis, ECP02-004, "Water Hammer Analysis - LPSI A," was performed specifically for Low Pressure Safety Injection Train A, again using Generic Letter 91-18 guidance to support an acceptable void size of 4 cubic feet. The team noted that the calculated acceptable void sizes were incorporated into Operating Procedure OP-903-026, "Emergency Core Cooling System Valve Lineup Verification," Revision 11. The team noted that guidance in this procedure stated that a new condition report was not required to be written provided void sizes below 4 cubic feet are found. The team also noted that since the last refueling outage approximately 22 occurrences were identified were Low Pressure Safety Injection Train A experienced voiding requiring the use of the Generic Letter 91-18 analysis to justify operability. The team noted that prior to the last refueling outage Low Pressure Safety Injection Train A also experienced extensive voided conditions requiring the Generic Letter 91-18 analysis to justify operability.

The team noted that the licensee had not considered use of this analysis a Generic Letter 91-18 issue that required timely corrective actions. Based on this the staff did not review this condition as is typically performed during each forced and refueling outage for all flagged Generic Letter 91-18 issues. The team determined that the licensee had failed to implement timely corrective actions to restore and operate the system in accordance with the design criteria of the system as described in the Final Safety Analysis Report.

<u>Analysis</u>: Using the guidance in Appendix B of Inspection Manual Chapter 0612, this issue screens more than minor because the Mitigating Systems Objective to ensure the availability, reliability, and capability is potentially affected when the system is maintained outside of its design criteria as described in the Final Safety Analysis Report. Using the Significance Determination Process Phase 1 screening worksheet, found in Appendix A of Inspection Manual Chapter 0609, the finding is of very low safety significance, Green, since the analysis used to assess the degraded condition ensured the system could perform its safety function.

<u>Enforcement</u>: 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," states, in part, that "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected." The failure to promptly implement timely corrective actions resulting in voiding conditions adversely affecting the design basis of the low pressure safety injection system since 1997, is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. The team determined this to be the second example of a Criterion XVI violation resulting from the failure to implement timely corrective actions to resolve a degraded condition. This violation is being treated as a noncited violation (NCV 50-382/0205-02) consistent with Section VI.A of the NRC Enforcement Policy. The licensee documented this issue in their corrective action process as Condition Report 2002-1356.

Example 3

Introduction: The licensee failed to maintain design control of the low pressure safety injection system, Train A, in accordance with the design basis as described in the Final Safety Analysis Report, when installing a modification to mitigate adverse voiding conditions that have affected the system. The failure to maintain design control of the system resulted in a potential loss of a Seismic Class 1, ASME Section III, Safety Class 2, barrier during post-accident conditions. The failure to maintain design control of the system is a violation of 10 CFR Part 50, Appendix B, Criterion III. The issue was characterized under the significance determination process as having very low safety significance.

Description: As previously discussed, the team reviewed corrective actions the licensee had taken to reduce intrusion of nitrogen into the low pressure safety injection system. These actions included replacement of Safety Injection Valve SI-142A during their previous refueling outage, and installation of automatic vents at Valves SI-133A and SI-1402A. The team noted that voids in Low Pressure Safety Injection A system were still being found, however, the frequency and size of the voids have been significantly reduced as a result of the licensee's corrective actions.

The team reviewed modification package (ER-W3-2002-0352-000) that installed the automatic vent valve in Low Pressure Safety Injection Train A. The modification was installed in June of 2002. The original design of the system is Seismic Class 1, ASME Section III, Safety Class 2. The modification consisted of installing two Safety Class 2 solenoid isolation boundary valves (SI-6011 and SI-6012) and a non safety automatic vent valve in series. During normal operation, gasses at Valve SI-133A migrate through the normally open solenoid operated isolation valves and enter the automatic vent that would discharge the gasses to the wing area of the reactor auxiliary building. The two solenoid operated valves were designed to close following a safety injection actuation signal to isolate the non safety related automatic vent from the system, thereby, preventing a leakage pathway from existing. The team noted Operations Procedure OP-902-002, "Loss of Coolant Accident Recovery," Revision 9, allows operators to reset the safety injection actuation signal following accident conditions when safety injection is no longer required. The team questioned whether or not the isolation valves for the automatic vent would reopen following reset of the safety injection signal resulting in a potential leakage pathway. Upon review of the design package the licensee determined that the boundary isolation valves would reopen resulting in the system being outside of its design basis configuration during post-accident conditions. The licensee implemented immediate corrective actions to change Operations Procedure OP-902-002, "Loss of Coolant Accident Recovery," directing operators to manually close Safety Injection Valves SI-6011 and SI-6012 prior to resetting the safety injection actuation signal. The team determined that the failure to maintain the design control of the Low Pressure Safety Injection Train A system in accordance with the design basis as described in the Final Safety Analysis Report was a violation of 10 CFR Part 50, Appendix B, Criterion III.

Analysis: Using the guidance in Appendix B of Inspection Manual Chapter 0612, this issue screens more than minor because the Barrier Integrity Objective to provide reasonable assurance that the physical design barriers protect the public from radionuclide releases caused by accidents or events was potentially affected when the system was modified. Because emergency operating procedures failed to keep Safety Injection Valves SI-6011 and SI-6012 closed when resetting the safety injection signal, this modification failed to maintain the Seismic Class 1, ASME Section III, Safety Class 2 design criteria for the low pressure safety injection system during post accident conditions as described in the Final Safety Analysis Report. Using the Significance Determination Process Phase 1 screening worksheet, found in Appendix A of Inspection Manual Chapter 0609, the finding was determined to be of very low safety significance (Green). This conclusion was based on the following facts: (1) the condition did not affect the mitigating system function and only affected the barrier integrity cornerstone, (2) no design bases accident had taken place while the plant was in this condition, (3) had a design bases accident occurred while this condition existed, a small leak of coolant would have occurred with minimal dose impact on both site personnel and the public, and (4) this small leak of coolant could have easily been identified and corrected by operators in the control room.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that "Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions." The failure to maintain design control of the Low Pressure Safety Injection Train A system resulting in loss of a Seismic Class 1, ASME Section III, Safety Class 2, barrier during post accident conditions, is a violation of 10 CFR Part 50, Appendix B, Criterion III. This violation is being treated as a noncited violation (NCV 50-382/0205-03) consistent with Section VI.A of the NRC Enforcement Policy. The licensee documented this issue in their corrective action process as Condition Report 2002-2042.

d. Assessment of Safety-Conscience Work Environment

(1) Inspection Scope

The inspectors interviewed approximately 12 individuals form the licensee's staff, which represented a cross-section of functional organizations and supervisory and nonsupervisory personnel. These interviews assessed whether conditions existed that would challenge the establishment of a safety-conscience work environment. The team also reviewed select concerns placed into the licensee's employee concerns program, which provides an alternate method to the corrective action program for employees to raise safety concerns with the option of remaining anonymous.

(2) Issues

The team identified no findings related to the safety-conscience work environment at the facility. The inspectors concluded, based on information collected and reviewed from the interviews, that employee's were willing to identify safety issues and enter them into a corrective action system.

4OA3 Event Followup

(Closed) Licensee Event Report 50-382/2002-006-00: Section 4OA2.1 describes the circumstances and licensee actions regarding operation in excess of 100 percent licensed power limit due to instrumentation biases.

4OA6 Exit Meeting

The team discussed the findings with Mr. Joseph Venable, Vice President Operations, and other members of the licensee's staff on December 20, 2002 and January 31, 2003. Licensee management did not identify any materials examined during the inspection as proprietary.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

S. Anders, Superintendent, Plant Security
K. Boodry, NSSS System Engineer
T. Brumfield, Manager, Quality Assurance
R. Conigliaro, System Engineer
C. DiMarco, Quality Assurance Specialist
R. Dodds, Vice President Technical Assistant
J. R. Douet, General Manager, Plant Operations
C. Fugate, Assistant Manager, Operations
T. Gaudet, Director, Planning and Scheduling
R. Gilmore, Mechanical Supervisor
B. Houston, Superintendent, Radiation Protection
J. Johnstone, Operating Experience Coordinator
C. Lambert, Director, Engineering
M. Langan, Corrective Action and Auditing Coordinator
J. Lewis, Emergency Planning Manager
D. Miller, ALARA Supervisor
R. Murillo, Senior Staff Licensing Engineer
R. Osborne, Manager, System Engineering
K. Peters, Director, Nuclear Safety Assurance/Emergency Preparedness
R. Peters, Six Sigma-Black Belt
G. Pierce, Chemistry Superintendent
O. Pipkins, Senior Licensing Engineer
B. Porter, Manager, Maintenance
J. Rachal, Engineering Supervisor
J. Reese, Manager, Design Engineering
J. A. Ridgel, Manager, Corrective Actions
G. Scott, Engineer, Licensing
T. E. Tankersley, Manager, Training
S. Trevillion, Supervisor of Procurement
J. Venable, Vice President, Operations
K. T. Walsh, Manager, Operations
R. Williams, Licensing Engineer
G. Zetsch, Senior Lead Coordinator, Security

NRC

A. Gody, Chief, Operations Branch
G. Larkin, Resident Inspector
L. Ricketson, Senior Health Physicist
T. Jackson, Resident Inspector

ITEMS OPENED AND CLOSED

Opened and Closed

50-382/02-05-01	NCV	Ineffective corrective actions resulting from inadequate evaluations of extent of condition
50-382/02-05-02	NCV	Ineffective corrective actions resulting from untimeliness
50-382/02-05-03	NCV	Failure to maintain design control of the low pressure safety injection system

Closed

50-382/2002-006-00	LER	Power in excess of 100 percent

DOCUMENTS REVIEWED

PLANT PROCEDURES

Document	Title	Revision
EPP 451	Emergency Planning Action Item Tracking System	1
HP 002-101	Dosimetry Investigation Reports	10
LI-102	Corrective Action Process	0,1,2,2W1
OE-100	Operating Experience Program	0,1
OP-903-001	Technical Specification Surveillance Logs	25
UNT-005-036	Reactivity Management Program	2
UNT-006-018	Condition Report Trending	6
UNT-007-027	Control of Boric Acid Corrosion on the Reactor Coolant System	3

DRAWINGS

D72-08300-710, “Schematic-S.V.S. Regulator 3 Phase With Paralleling,” Portec, Inc., Revision F.

D72-11500-710, Sheet 2 of 3, “System Schematic-Regulator,” Portec, Inc., Revision E.

CONDITION REPORTS

CR-WF3-1996-00312
CR-WF3-1996-01239
CR-WF3-1996-01299
CR-WF3-1997-01047
CR-WF3-1998-00212
CR-WF3-1999-01210
CR-WF3-1999-01211
CR-WF3-2000-00199
CR-WF3-2000-00313
CR-WF3-2000-00351
CR-WF3-2000-00460
CR-WF3-2000-00994
CR-WF3-2000-01226M
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CR-WF3-2001-0059
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CR-WF3-2002-00079
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CR-WF3-2002-00136
CR-WF3-2002-00150
CR-WF3-2002-00163
CR-WF3-2002-00168
CR-WF3-2002-00169
CR-WF3-2002-00175
CR-WF3-2002-00180
CR-WF3-2002-00184
CR-WF3-2002-00189
CR-WF3-2002-00190
CR-WF3-2002-00211
CR-WF3-2002-00215
CR-WF3-2002-00300
CR-WF3-2002-00301
CR-WF3-2002-00379
CR-WF3-2002-00106
CR-WF3-2002-00126
CR-WF3-2002-00322
CR-WF3-2002-00328
CR-WF3-2002-00335
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CR-WF3-2002-00346
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CR-WF3-2002-00914
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CR-WF3-2002-00954
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CR-WF3-2002-00839	CR-WF3-2002-01027	CR-WF3-2002-01144
CR-WF3-2002-00959	CR-WF3-2002-01032	CR-WF3-2002-01145
CR-WF3-2002-00863	CR-WF3-2002-01033	CR-WF3-2002-01179
CR-WF3-2002-00873	CR-WF3-2002-01057	CR-WF3-2002-01202
CR-WF3-2002-00884	CR-WF3-2002-01059	CR-WF3-2002-01242
CR-WF3-2002-00890	CR-WF3-2002-01069	CR-WF3-2002-01255
CR-WF3-2002-00960	CR-WF3-2002-01078	CR-WF3-2002-01351
CR-WF3-2002-00981	CR-WF3-2002-01079	CR-WF3-2002-01352
CR-WF3-2002-00983	CR-WF3-2002-01081	CR-WF3-2002-01353
CR-WF3-2002-00986	CR-WF3-2002-01088	CR-WF3-2002-01355
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CR-WF3-2002-01011	CR-WF3-2002-01114	CR-WF3-2002-01710
CR-WF3-2002-01021	CR-WF3-2002-01115	CR-WF3-2002-01915
CR-WF3-2002-01025	CR-WF3-2002-01139	

SIGNIFICANT CONDITION REPORTS

CR-WF3-2001-00782	CR-WF3-2002-00168	CR-WF3-2002-00628
CR-WF3-2001-01102	CR-WF3-2002-00169	CR-WF3-2002-00726
CR-WF3-2001-01225	CR-WF3-2002-00175	CR-WF3-2002-00775
CR-WF3-2001-01284	CR-WF3-2002-00215	CR-WF3-2002-00818
CR-WF3-2001-01295	CR-WF3-2002-00382	CR-WF3-2002-00983
CR-WF3-2001-01301	CR-WF3-2002-00465	CR-WF3-2002-01242
CR-WF3-2001-01399	CR-WF3-2002-00468	CR-WF3-2002-01312
CR-WF3-2002-00041	CR-WF3-2002-00476	CR-WF3-2002-01529
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CALCULATIONS

EC-P97-001, “Operability Evaluation-LPSI B, “Revision 0, June 17, 1997
EC-P97-002, “Operability Evaluation-LPSI B, “Revision 0, June 17, 1997
EC-P97-003, “Operability Evaluation-LPSI B, “Revision 0, June 17, 1997
EC-M97-002, “Water Hammer Analysis - LPSI B,” Revision 0, June 17, 1997
EC-M97-003, “Water Hammer Analysis - LPSI A,” Revision 0, May 16, 1997
ECP02-004, “Water Hammer Analysis - LPSI A,” Revision 0, August 21, 2002

AUDITS AND ASSESSMENTS

W3H3-2001-0041, “Quality Assurance Audit of Corrective Action Program,” March 20, 2001

“Assessment Report - Root Cause Analysis Process and Culture,” September 4, 2001

WL0-2002-028, “Snapshot Assessment on Effectiveness Review for CR-WF3-2000-0372 - Ineffective Corrective Action”

CR-WLO-2002-00075, “Snapshot Assessment - Focus of Corrective Actions For Significant Condition Reports”

CR-WLO-2002-085, “Corrective Action / Condition Report Closure Process Assessment,” August 12 – 16, 2002

CR-WLO-2002-088, “Snap Shot Assessment on Condition Reports Closed to Maintenance Action Items,” August 16, 2002

LO-WLO-2002-00048, “Design Engineering Assessment - RF-11 ERs,” July 11, 2002

LO-WLO-2001-0001 CA 81, “Heat Exchanger Assessment,” June 6, 2002

“Snapshot Assessment on External Corrosion Program”

“Radiation Monitoring System Assessment,” November 14, 2002

“Snapshot Assessment on the Power Supply Program,” September 13, 2002

CR-WF3-WLO-2002-114 CA#1, “MOV Self Assessment Report,” November 22, 2002

“Engineering Department Assessment,” November 8, 2002

LO-WLO-2001-00170 CA# 22, “Snapshot Assessment on Configuration Control Process Changes
Resulting from INPO Evaluation Area for Improvement,” October 30, 2002

Computer Software & Communications Security Assessment (March 1 - August 30, 2002)

Engineering Human Performance Quarterly Assessment (February - April 2002)

Engineering Human Performance Quarterly Assessment (May - July 2002)

Engineering Human Performance Quarterly Assessment (August - October 2002)

<u>Learning Organization Operating Experience (LO-OPX)</u>

2001-0006	2001-00204	2002-00012	2002-00069
2001-0014	2002-00003	2002-00013	2002-00078
2001-0019	2002-00007	2002-00014	2002-00089
2001-00041	2002-00010	2002-00025	2002-00221
2001-00071	2002-00011	2002-00027	2002-235

<u>LETTERS</u>

W3F1-2001-0081	W3F1-2002-0037
W3F1-2001-0104	W3F1-2002-0051
W3F1-2002-0032	

OTHER

Dosimetry Incident Reports (2002)

Emergency Planning Action Items (Open and closed - 2002)

Entergy Licensing Position 2, “Evaluation and Resolution of Degraded and Nonconforming Conditions,” January 5, 2001

Monthly Radiation Protection Report (December 2001, October 2002)

Problem Statement: Core Protection Calculators (CPCs),” December 3, 2002

Safeguard Event Logs (Second quarter 2001 - Fourth quarter 2002)

W3P90-0226, “Summary of Actions in Progress for NRC Information Notice 88-70 Check Valve Inservice Testing Program Deficiencies,” January 25, 1990

W3F1-2001-0081, “30 Day Response to NRC Bulletin 2001-01 for 2001-01 for Waterford 3; Circumferential Cracking of VHP Nozzles,” September 4, 2001

W3F1-2001-0104, “Supplement to 30 Day Response to NRC Bulletin 2001-01 for 2001-01 for Waterford 3; Circumferential Cracking of VHP Nozzles,” November 8, 2001

W3F1-2002-0032, “15 Day Response to NRC Bulletin 2002-01 for Waterford 3; Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,” April 1, 2002

W3F1-2002-0037, “30 Day Response to NRC Bulletins 2001-01 and 2002-01 for Vessel Head Inspection Findings,” April 16, 2002

W3F1-2002-0051, “60 Day Response to Bulletin 2002-01 - Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,” May 16, 2002

W3 Quarterly Trend Report - 3rd Quarter 2002

INITIAL MATERIAL REQUESTED

- All procedures governing or applying to the corrective action program, including the processing of information regarding generic communications and industry operating experiences.

- Procedures and descriptions of any informal systems, especially used by operations, for issues below the threshold of the formal corrective action program.

- Index of all corrective action documents (i.e., condition reports) that were initiated or closed from May 2001 to November 2002.

- All major corrective action documents (i.e., those that roll-up one or more smaller issues) from May 2001 to November 2002.

- List of all corrective action documents categorized as “Significant Conditions” requiring root cause analysis from May 2001 to November 2002.

- All corrective action documents associated with nonescalated no response required or noncited violations from May 2001 to November 2002.

- All corrective action documents associated with voiding conditions found in safety related systems from May 2001 to November 2002.

- All corrective action documents associated with the ultrasonic flow monitoring system from May 2001 to November 2002.

- List of all self assessments or QA assessments/audits from May 2001 to November 2002.

- Control room logs from May 2001 to November 2002.

- List of all MAI’s (maintenance action items) from May 2001 to November 2002.

- Procedures associated with the control of boric acid corrosion on the reactor coolant system, and all corrective action documents or MAI’s written to address boric acid corrosion (such as leaking valves, etc...).

- Procedures for the check valve monitoring, maintenance and trending program, and all corrective action documents and MAI’s associated with check valves in safety related systems.

- All corrective action program reports or metrics used for tracking effectiveness of the corrective action program from May 2001 to November 2002.

- All quality assurance audits and surveillances, and functional self assessments of corrective action activities completed from May 2001 to November 2002.

- Safeguards event logs from May 2001 to November 2002.

- Radiation protection event logs from May 2001 to November 2002.

- List of all SSC’s placed in or removed from the maintenance rule a(1) category from May 2001 to November 2002.

- All corrective action documents (from May 2001 to November 2002) associated with:

  (1) Repetitive problems or issues
  (2) Ineffective corrective actions/Untimely corrective actions

- List of all corrective action documents (from May 2001 to November 2002.) associated with:

(1) Human performance issues
(2) Occupational and general public exposure issues
(3) Emergency preparedness issues
(4) Security

- All corrective action documents related to the following industry operating experience generic communications:

  Part 21 Reports:

  Report 2001-27-0 Flow Serve, Excessive disc angular movement in swing check valves

  Report 2001-30-0 Flow Serve, Galling and binding of safety-related containment isolation valves

  Report 2001-35-0 Dresser-Rand, Material misapplication in terry turbine trip and throttle valve screw spindles

  Report 2002-03-0 ASCO Valve Hydramotor® actuator limitswitch

  Report 2002-13-0 Solid State Controls, Potential for loose fan blade in uninterruptible power systems

  NRC Generic Letter

  Generic Letter 88-05: Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants

  NRC Information Notices:

  NRC Information Notice 88-70: Check Valve Inservice Testing Program Deficiencies

  NRC Information Notice 2001-06: Centrifugal Charging Pump Thrust Bearing Damage Not Detected Due to Inadequate Assessment of Oil Analysis Results and Selection of Pump Surveillance Points

  NRC Information Notice 2001-14: Problems with Incorrectly-installed Swing-check Valves

  NRC Information Notice 2001-19: Improper Maintenance and Reassembly of Automatic Oil Bubblers

  NRC Information Notice 2002-03: Highly Radioactive Particle Control Problems During Spent Fuel Pool Cleanout

  NRC Information Notice 2002-10: Nonconservative Water Level Setpoints on Steam Generators

  NRC Information Notice 2002-11: Recent Experience with Degradation of Reactor Pressure Vessel Head

NRC Information Notice 2002-12: Submerged Safety-related Electrical Cables

NRC Information Notice 2002-13: Recent Experience with Degradation of Reactor Pressure Vessel Head

NRC Information Notice 2002-14: Ensuring a Capability to Evacuate Individuals, Including Members of the Public, from the Owner-Controlled Area

NRC Information Notice 2002-25: Challenges to Licensees Ability to Provide Prompt Public Notification and Information During an Emergency Preparedness Event

NRC Information Notice 2002-27: Recent Fires at Commercial Nuclear Power Plants in the United States