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

August 5, 2005

Joseph E. Venable
Vice President Operations
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Entergy Operations, Inc.
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SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000382/2005003

Dear Mr. Venable:

On June 26, 2005, the NRC completed an inspection at your Waterford Steam Electric Station, Unit 3. The enclosed report documents the inspection findings which were discussed on July 6, 2005, with Mr. K. Walsh, General Plant Manager, and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

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

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

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

Sincerely,

/RA/

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

Docket: 50-382
License: NPF-38

Enclosure:
NRC Inspection Report
050000382/2005003
w/attachment: Supplemental Information

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C:DRS/EB1	C:DRS/PSB	C:DRS/OB	C:DRS/EB2	C:DRP/E
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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-382
License No.:	NPF-38
Report No.:	50-382/05-003
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	April 8 through June 26, 2005
Inspectors:	M. C. Hay, Senior Resident Inspector G. F. Larkin, Resident Inspector R. Azua, Project Engineer V. Gaddy, Senior Project Engineer J. Kirkland, Project Engineer D. R. Carter, Health Physicist, Plant Support Branch B. D. Baca, Health Physicist, Plant Support Branch J. P. Reynoso, P.E., Reactor Engineer, Engineering Branch B. K. Tharakan, Health Physicist, Plant Support Branch G. George, Reactor Inspector B. Tindell, Reactor Inspector
Accompanying Personnel:	L. Ellershaw, P.E., Consultant
Approved By:	D. N. Graves, Chief, Project Branch E
ATTACHMENTS:	Supplemental Information

Enclosure

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SUMMARY OF FINDINGS

IR05000382/2005-003; 04/08/05-06/26/05; Waterford Steam Electric Station, Unit 3; Integrated Resident and Regional Report; Surveillance Testing

The report covered a 11-week period of inspection by resident inspectors, a project engineer, a senior project engineer, three health physicists, a reactor engineer, and two reactor inspectors. The inspectors identified one Green finding. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, for the failure to maintain design control of the Train B emergency diesel fuel oil storage tank level instrument sensing line resulting in level indication error. This error affected the ability of Train B fuel oil storage tank to provide sufficient fuel oil to support 7 days of continuous diesel generator operations following a loss of offiste power and a design-bases accident.

  This finding was greater than minor because it affected the mitigating systems cornerstone objective of ensuring the capability of emergency power to respond to initiating events to prevent undesirable consequences. Since the finding represented an actual loss of safety function, for a single train, for greater than its Technical Specification-allowed outage time, the finding was analyzed using Phase 2 of the Significant Determination Process. The finding was of very low safety significance because the licensee staff would have sufficient time to order replacement fuel, procedures existed to order replacement fuel and training was conducted on the existing procedures under conditions similar to the initiating event assumed (Section 1R22).

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status: The plant began the period on April 8, 2005, at 100 percent power until April 17 when the plant was shutdown for Refueling Outage 13. On June 10 operators commenced a reactor startup to perform low power physics testing. Power was increased and reached approximately 100 percent on June 17. Power remained at that level for the remainder of the inspection period.

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

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors completed a review of Entergy's preparations for impending adverse weather conditions in relation to Tropical Storm Arlene, high winds and missile protection. The review included plant procedures, the Updated Final Safety Analysis Report (UFSAR), and the corrective action program to ensure adverse weather readiness of safety-related, risk significant systems. The inspectors performed multiple walkdowns of the turbine deck, wet and dry cooling towers, and the protected area to verify the licensee had identified and corrected conditions that could potentially impact safety equipment performance.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

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

- On April 20, 2005, the inspectors walked down the accessible portions of the spent fuel pool cooling system, Train A. The walkdown was completed prior to 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 16. The inspectors also witnessed makeup addition to the spent fuel pool from the condensate storage pool, in accordance with Operating Procedure OP-002-006, Section 6.9.

- On April 29, 2005, the inspectors walked down the mechanical and electrical components of a critical portion of shutdown cooling system Train B. The inspectors considered whether the system was properly aligned as described in the UFSAR and Technical Specifications. This inspection focused on verifying that system valve and electrical breaker alignments were appropriate and that system instrumentation was both available and functional. The walk down was conducted using Operating Procedure OP-009-005, "Shutdown Cooling," Revision 17.

- On May 9, 2005, the inspectors performed a partial equipment alignment inspection of emergency diesel generator Train B while emergency diesel generator Train A was inoperable. A review of select maintenance work orders and corrective action documents was performed to assess the material condition and performance of emergency diesel generator Train A. System configuration was assessed using Operating Procedure OP-009-002, “Emergency Diesel Generator,” Revision 18-4. 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.

1R05 Fire Protection (71111.05)

.1 Routine Fire Protection Inspections

a. Inspection Scope

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

The following areas were inspected:

- Fire Zone Reactor containment building on April 17, 2005
- Fire Zone RAB 1B, 1C, 1D, 8A, 8B, and 8C on April 21, 2005
- Fire Zone RAB 1A, 8C, 11, 12, 13, Roof E, and Roof W on April 29, 2005
- Fire Zone RAB 1A, 2, 32, and Cooling Tower A on May 1, 2005
- Fire Zone RAB 1A, 8A, 8B, 8C, and 15 on May 1, 2005
- Fire Zone Reactor containment building on May 25, 2005

b. Findings

No findings of significance were identified.

.2 Routine Fire Drill Inspection

a. Inspection Scope

The inspectors observed a site fire drill performed on June 18, 2005. The simulated fire was located in the supplementary chiller building. The inspectors assessed the fire brigade's performance in the following areas:

- Appropriate clothing donned in a timely manner
- Self-contained breathing apparatus properly worn and used
- Fire fighting preplan strategies were used
- The fire area was entered in a controlled manner
- Sufficient fire fighting equipment was brought to the scene
- Effective command and control provided by the fire brigade leader

The inspectors also reviewed the fire drill critique to verify that areas for improvement were properly identified and all the scenario objectives were met.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors performed a semiannual inspection of internal flood protection features in the component cooling water pump rooms. The rooms contain portions of both trains of component cooling water. The inspection included a review of the UFSAR, selected design calculations, Regulatory Guide 1.102, "Flood Protection for Nuclear Plants," Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," and a walkdown of flood protection features in the component cooling water pump rooms.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors completed a review of performance tests for the emergency diesel generator Trains A and B jacket water cooling system and lube oil cooling system. The review assessed the adequacy of the licensee's periodic maintenance method and implemtation of bio-fouling controls. The inspectors also conducted a walkdown of the heat exchangers and their associated system interfaces.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

Inspection Procedure 71111.08 requires a minimum sample size of four within sections (Sections 02.01, 02.02, 02.03, and 02.04).

02.01: Performance of Nondestructive Examination Activities Other Than Steam Generator Tube Inspections, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control

a. Inspection Scope

The inspection procedure requires the review of nondestructive examination activities consisting of two or three different types (i.e., volumetric, surface, or visual). The inspectors reviewed the radiographic examination (volumetric) records and liquid penetrant (surface) records of Field Welds 2A and 3A of replaced Safety Injection Valve 407A. The inspectors witnessed the performance of ultrasonic examination (volumetric) on four pressurizer heater nozzles, and ultrasonic examination (volumetric) and eddy-current examination (combination surface and volumetric) on eight reactor vessel upper head penetration nozzles, and ultrasonic examinations (volumetric) on four cold shutdown piping welds. The table below identifies the above 28 examinations, which were conducted using 4 methods in 2 different examination types.

Component	Identity	Examination Type	Examination Method
Reactor Vessel Upper Head Control Element Drive Mechanism Penetration Nozzles	Nozzles 29, 33, 34, and 37	Volumetric Combination Surface and Volumetric	Ultrasonic Eddy-Current
Reactor Vessel Upper Head In-Core Instrumentation Penetration Nozzles	Nozzles 97, 98, 99, and 100	Volumetric Combination Surface and Volumetric	Ultrasonic Eddy-Current
Pressurizer Heater Nozzles	Nozzles A4, F1, G3, and H1	Volumetric	Ultrasonic

Safety Injection Piping Welds Adjacent to and Sequentially Downstream from Safety Injection Valve SI-405B	FW 52-001, FW 52-002, FW 52-003, and FW 52-004	Volumetric	Ultrasonic
Safety Injection Valve SI 407A Welds: Valve to Elbow and Valve to Piping	FW-2A and FW- 3A	Volumetric Surface	Radiography Liquid Penetrant

For each of the nondestructive examination activities reviewed, the inspectors verified that the examinations were performed in accordance with site procedures and the applicable American Society of Mechanical Engineers (ASME Code) requirements.

During review of each examination, the inspectors verified that appropriate nondestructive examination procedures were used, that examinations and conditions were as specified in the procedure, and that test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors observed calibration of the Sonic-136 ultrasonic equipment system calibration performed on May 3, 2005, and the recheck performed on May 4, 2005. This equipment was used to perform the examinations on the safety injection piping welds identified in the table above. The inspectors also verified the nondestructive examination certifications of the two licensee personnel who performed the safety injection piping weld ultrasonic examinations, and the corresponding certifications of the six contractor personnel who performed the identified ultrasonic, eddy-current, liquid penetrant, and radiographic examinations. Finally, the inspectors observed that indications identified during the ultrasonic and eddy-current examinations were dispositioned in accordance with the ASME qualified nondestructive examination procedures used to perform the examinations.

The inspection procedure required review of one or two examinations with recordable indications that were accepted for continued service, to ensure that the disposition was made in accordance with the ASME Code. Discussion with licensee inservice inspection management personnel revealed that no relevant indications in ASME Code Section XI components have been accepted for continued service. The licensee's practice has been to remove all identified relevant indications.

The inspection procedure further required verification of one to three welds on Class 1 or 2 pressure boundary piping to ensure that the welding process and welding examinations were performed in accordance with the ASME Code. The inspectors observed welding performed on two pressurizer heater sleeves for Heaters G-3 and G-4. This welding was an automatic gas tungsten arc welding process used during the half nozzle repairs utilizing a midwall sleeve weld. The inspectors also reviewed the welding records of two field welds associated with the ASME Code, Section XI, replacement of Safety Injection Valve SI-407A.

The inspectors verified, by review, that the welding procedure specifications and the welders had been properly qualified in accordance with ASME Code, Section IX, requirements. The inspectors also verified, through observation and record review, that essential variables for the gas tungsten arc welding process had been identified and were monitored.

The inspectors completed one sample under this section.

b. Findings

No findings of significance were identified.

02.02: Reactor Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

The inspection requirements for this section parallel the inspection requirement steps in Section 02.01. The inspectors observed the 16 nondestructive examinations on the eight reactor vessel upper head penetrations identified in the table.

Additionally, the nondestructive examination procedures used to perform the above examinations were reviewed to assure that they were consistent with ASME Code requirements, and the equipment and calibration requirements were appropriately identified and demonstrated. The inspectors also reviewed the in-core instrumentation nozzle mockup used to qualify the equipment and procedure demonstration qualification. The nondestructive examination records were also reviewed to verify that 100 percent inspection coverage was achieved on the observed penetration nozzles.

The inspectors verified that the nondestructive activities were performed in accordance with the requirements of NRC Order EA-03-009, “Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors.”

The nondestructive examinations performed during the NRC inspection did not reveal any defects. Indications were dispositioned in accordance with the licensee’s qualified procedures and in accordance with ASME Code acceptance criteria parameters.

The inspectors determined through discussions with licensee personnel, that welding repairs have not been performed on upper head penetrations.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

02.03: Boric Acid Corrosion Control Inspection Activities (Pressurized Water Reactors)

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's boric acid corrosion control program for monitoring degradation of the reactor coolant system, emergency core cooling system, chemical and volume control system, and the containment spray systems.

The inspection procedure requires review of a sample of boric acid corrosion control walkdown visual examination activities through either direct observation or record review. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walkdown during Mode 3 as specified in Procedure UNT-007-027. Visual records of the components and equipment were also reviewed by the inspectors.

Additionally, the inspectors independently performed examinations of piping containing boric acid during a walkdown of the containment building, the reactor auxiliary building wing areas, and the safeguards pump rooms.

The inspection procedure requires verification that visual inspections emphasize locations where boric acid leaks can cause degradation of safety significant components. The inspectors verified through direct observation and program/record review that the licensee's boric acid corrosion control inspection efforts are directed towards locations where boric acid leaks can cause degradation of safety-related components.

The inspection procedure requires a review of one to three engineering evaluations performed for boric acid leaks found on reactor coolant system piping and components. The inspectors reviewed Engineering Evaluations 05-0176, 05-0177, and 05-0178, which addressed boric acid leaks identified on pump seals and upper seal flanges on three of the four reactor coolant pumps. The evaluation appropriately addressed the causes and corrective actions, and included an assessment of identified corrosion damage. This included assurance that ASME Code minimum wall thickness requirements have been maintained.

The inspectors verified that the conditions were identified in the licensee's corrective action program in Condition Report CR-WF3-2005-01361.

Finally, the inspection procedure requires a review of one to three corrective actions performed for identified boric acid leaks. The inspectors conducted a review of the licensee's boric acid corrective actions identified since March 2002. The condition reports are identified in the Attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

02.04: Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspection procedure specified performance of an assessment of in-situ screening criteria to assure consistency between assumed nondestructive examination 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.

At the time of this inspection, no conditions had been identified that warranted in-situ pressure testing. The inspectors did, however, review the licensee's report, "Waterford Steam Generator Degradation Assessment for RF13," dated April 23, 2005, and compared the in-situ test screening parameters to the guidelines contained in the EPRI document "In-Situ Pressure Test Guidelines," Revision 2. This review determined that the screening parameters were consistent with the EPRI guidelines.

In addition, the inspectors reviewed both the licensee site-validated and qualified acquisition and analysis technique sheets used during this refueling outage; and the qualifying EPRI examination technique specification sheets to verify that the essential variables regarding flaw sizing accuracy, tubing, equipment, technique, and analysis had been identified and qualified through demonstration. The inspector-reviewed acquisition technique and analysis technique sheets are identified in the Attachment.

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 the licensee's prediction capability. The inspectors compared the previous outage operational assessment predictions with the flaws identified thus far during the current steam generator tube inspection effort. Compared to the projected damage mechanisms identified by the licensee, the number of identified indications fell within the range of prediction and were quite consistent with predictions. The number of top of tubesheet axial indications, however, was lower than predicted. The inspectors determined that the flaw degradation severity levels found, thus far, were well within the predicted expectations. It was also determined that the number of tubes identified for plugging during this outage were consistent with the predicted number. During the inspection, the resolution analysts had discovered an unusual wear and batwing anomaly at BW9 location in Steam Generator 32 using a plus point eddy-current examination probe. The licensee was still performing assessment of this condition at the end of this inspection.

The inspection procedure specified confirmation that the steam generator tube eddy-current test scope and expansion criteria meet Technical Specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors evaluated the recommended steam generator tube eddy-current test scope established by Technical Specification requirements and the Waterford, Unit 3, degradation assessment report, which incorporated inputs from Waterford's condition monitoring report and operational assessment. The data was compiled and documented in a section of the Unit 3 degradation assessment. The inspectors compared the recommended test scope to the actual test scope and found that the licensee had expanded their inspection because of new wear indications. This increased inspection scope included a 100 percent plus point inspection of all wear including the new wear indications found in the batwing region. The inspectors were made aware that the licensee had placed the steam generators in Technical Specification C-3 category based on the total number of defective tubes identified during eddy-current testing. These results were documented in the licensee’s corrective action program (Condition Report CR-WF3-2005-01918).

The inspection procedure specified, if new degradation mechanisms were identified, verification that the licensee fully enveloped the problem in its analysis of extended conditions including operating concerns, and had taken appropriate corrective actions before plant startup. The eddy-current test results, to date, had not identified any new degradation mechanisms.

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

The inspection procedure further required verification that repair processes being used were approved in the Technical Specifications. During this inspection, the inspectors observed the installation of mechanically rolled plugs: two in the hot-leg side and one in the cold-leg side of Steam Generator 31, one in the hot-leg side and five in the cold-leg side of Steam Generator 32. At the time of this inspection, it was estimated that a total of approximately 163 tubes would be plugged. The inspectors verified that this particular plugging operation was an NRC approved repair process.

The inspection procedure also required confirmation of adherence to the Technical Specification plugging limit, unless alternate repair criteria have been approved. The inspection procedure further required 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 determined that the Technical Specification plugging limits were being adhered to (i.e., 40 percent maximum through-wall indication).

If steam generator leakage greater than 3 gallons per day was identified during operations or during post shutdown visual inspections of the tubesheet face, the inspection procedure required verification that the licensee had identified a reasonable cause based on inspection results and that corrective actions were taken or planned to address the cause for the leakage. The inspectors did not conduct any assessment because this condition did not exist.

The inspection procedure required confirmation that the eddy-current test probes and equipment were qualified for the expected types of tube degradation and an assessment of the site-specific qualification of one or more techniques. The inspectors observed portions of eddy-current test performed on the following locations in Steam Generators 31 and 32: full length, U-bends, hot-leg square bends, and special interest locations. 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 inspectors performed a review of site-specific qualifications of the techniques being used. These are identified in the Attachment.

If loose parts or foreign material on the secondary side were identified, the inspection procedure specified confirmation that the licensee had taken or planned appropriate repairs of affected steam generator tubes, and that they inspected the secondary side to either remove the accessible foreign objects, or performed an evaluation of the potential effects of inaccessible object migration and tube fretting damage.

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

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On June 23, 2005, the inspectors observed a licensed operator simulator training scenario. During the scenario, operators responded to problems associated with multiple instrument failures, a main feedwater pump trip, a reactor cutback, and a dropped control rod. The simulator training evaluated the operators' ability to recognize, diagnose, and respond to abnormal and emergency reactor plant conditions. The inspectors observed and evaluated the following areas:

- Understanding and interpreting annunciator and alarm signals

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

- Use and adherence of Technical Specifications

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

- The crew's and evaluator's critiques

- Classifying emergencies and making notifications

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

During the inspection period, the inspectors reviewed licensee 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 the licensee's implementation of the Maintenance Rule to the requirements outlined in 10 CFR 50.65, and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2. The inspectors reviewed the following two components and/or systems that displayed performance problems:

- Main steam system
- 125 Vdc Batteries

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- On April 4, 2005, for removal of reactor power cutback to support circulating water Box C2 maintenance

- On April 8, 2005, for Refueling Outage 13 overall outage risk assessment

- On April 13, 2005, for planned maintenance on Startup Transformer A

- On May 16-17, during reactor coolant system reduced inventory configuration

b. Findings

No findings of significance were identified.

1R14 Non-Routine Evolutions and Events (71111.14)

a. Inspection Scope

On May 22, 2004, the inspectors responded to an inadvertent void that had been formed in the reactor vessel head during cold shutdown conditions. The inspectors reviewed control room logs, system parameter trends, and interviewed control room operators to assess plant conditions. Additionally, the inspectors remained in the control room during recovery actions to verify proper cooling was maintained by the shutdown cooling system.

b. Findings

No findings were identified related to the licensee identifying and correcting this adverse condition. The circumstances that created the voided condition were evaluated during a special inspection with findings and observations that will be documented in NRC Inspection Report 05000382/200510.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- Operability evaluation addressing Valve PMU-151, the PMU Containment Isolation Valve and its associated remote position indication (Condition Report CR-WF3-2005-1900)

- Operability evaluation addressing Main Steam Isolation Valve No. 1 actuator piston lower seal degradation (Condition Report CR-WF3-2005-1836)

- Operability evaluation addressing component cooling water Valve CC-413A failing open (Condition Report CR-WF3-2005-1279)

- Operability evaluation addressing the Steam Generator Number 1 hot leg nozzle dam following the failure of one of its pins to fully engage (Condition Report CR-WF3-2005-01592)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors completed one evaluation of the effect of one operator workaround. The inspectors also reviewed the second quarter 2004 Watchstation Deficiency List and assessed the effect of the workarounds on the ability of operators to implement plant emergency operating procedures. The inspectors completed the review to verify that the cumulative effect of workarounds did not challenge the operators' capability to respond to plant transients and events. The inspectors completed an in-office review, control room walkdown, and discussed with operators a workaround involving failure of Unit Auxiliary Transformer 3B fan motors to control in automatic. As a compensatory measure the fans were operated in manual requiring increased monitoring.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed a permanent plant modification to the atmospheric dump valves to ensure that the modification did not adversely affect system operability or design requirements specified in the UFSAR and Technical Specifications. The atmospheric dump valve setpoint controller was modified in support of the extended power uprate. The inspectors reviewed the following documentation during this inspection activity:

- Engineering Request W3-2003-0280-000, Revision 0, "Revise Atmospheric Dump Valve Control Loop"
- ANSI N45.2.4-1972, "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"
- License Amendment 198 to NPF-38, "Extended Power Uprate"

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

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

- Startup transformer Train A, following maintenance on April 17, 2005
- Charging Pump A/B, following maintenance on April 17, 2005
- Low steam generator pressure trip bistables and annunciator setpoints on January 26, 2005
- Emergency diesel generator fuel oil transfer Pump B, following emergent repairs on May 30, 2005
- Charging Pump A, following emergent repairs on May 13, 2005

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

Refueling Outage 13 started on April 17, 2005 and ended on June 11, 2005. During the outage, the inspectors observed shutdown, cooldown, refueling, startup, and maintenance activities to verify that the licensee maintained plant capabilities within the applicable Technical Specifications requirements and within the scope of the outage risk plan. Specific performance activities evaluated include:

- Clearance Activities - to ensure tags were properly hung and equipment appropriately configured to support the function of the clearance
- Reactor Water Inventory Controls - to verify flow paths, equipment configurations, and alternative means for inventory addition were appropriate to prevent inventory loss
- Reactivity Controls - to ensure compliance with Technical Specifications and to

verify activities, which could affect reactivity, were reviewed for proper control within the outage risk plan

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

- Reduced Inventory and Midloop Conditions - to verify 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 - to verify that operating parameters were established and maintained within the required range

- Reactor Coolant System Instrumentation Indication - to verity 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 - to ensure 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 four surveillance tests to ensure the systems were capable of performing their safety function and to assess their operational readiness. Specifically, the inspectors considered whether the following surveillance tests met Technical Specifications, the UFSAR, and licensee procedural requirements:

- Surveillance Procedure OP-903-024, “Reactor Coolant System Water Inventory Balance,” Revision 13, performed on March 7, 2005. This surveillance is used to determine the rate of identified and unidentified reactor coolant system leakage during steady-state operations. In addition, the inspectors evaluated Entergy's accumulated results for both the months of February and March 2005 against the criteria set forth in Manual Chapter 2515, Appendix D, Attachment 1, "Assessing Reactor Coolant System (RCS) Unidentified Leakage Rate Trend." Trends that resulted in Entergy entering one or more of the actions levels were found to have been appropriately evaluated and dispositioned.

- Administrative Procedure ME-001-012, "Temporary Power From Temporary Diesel for 3A2 and 3B2 4kV Buses (Modes 1-6)," Revision 1, and Operating Procedure OP-TEM-008, "Emergency Diesel Generator A(B) Backup Temporary Diesel Generator(s)" were used to test the Temporary Diesels 3A2 and 3B2 following their installation on February 23, 2005.
- Calibration Procedure MI 005-205, "Calibration/Functional Test of Level Instruments," Revision 8, performed on June 16, 2005. This procedure is used to calibrate and functionally test safety related level instruments.

- Procedure NOECP-253, "ASME Section XI Pressure Test," Revision 4, performed on June 9, 2005. This procedure performed a walkdown of the reactor containment building at normal operating pressure and temperature to identify boric acid leakage.

b. Findings

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, for the failure to maintain design control of the Train B emergency diesel fuel oil storage tank level instrument sensing line resulting in level indication error. This error affected the ability of Train B fuel oil storage tank to provide sufficient fuel oil to support 7 days of continuous diesel generator operations following a loss of offsite power and a design-bases accident.

Description. On June 15, 2005, the control room received an annunciator for low level in diesel fuel oil storage Tank B. Maintenance was sent to troubleshoot the problem and discovered that air was trapped in the instrument line to level instrument EGFILI6994-1B. After venting the line, maintenance discovered that level indication dropped from 98.3 percent to 96.8 percent. Technical Specification 3/4.8.1.1 and 3/4.8.1.2 require:

G. A minimum of 39,300 gallons (97.9 percent) of fuel, or

H. A fuel oil volume less than 39,300 gallons (97.9 gallons) and greater than 37,300 gallons (92.3 percent) of fuel for a period not to exceed 5 days (provided replacement fuel is onsite within the first 48 hours).

The inspectors noted the last manipulation of level instrument, EGFILI6994-1B, was on May 5, 2005, for performance of Calibration Procedure MI 005-205, "Calibration/Functional Test of Level Instruments." Section 9.7.5 of this procedure required venting the instrument during restoration. Upon questioning the inspectors were told that the instrument was vented in accordance with procedural guidance.

The inspectors reviewed maintenance records, operator logs, and corrective action documents and noted that on March 16, 2005, the licensee had initiated a condition report after noting approximately a 1 percent change in Train B fuel oil storage tank level over a one day period with no evolutions taking place that would have changed the level. The licensee determined that the instrument was most likely not adequately vented following a drain and fill of the fuel oil storage tank that took place February 28 through March 6, 2005.

Based on multiple occurrences of indicated level abnormalities the inspectors performed a walkdown of both Trains A and B diesel fuel oil storage systems. The inspectors noted that the sensing line for level instrument EGFILI6994-1B was not continuously sloped from the instrument to the process pipe as required by “Waterford 3 Instrument Installation Details” design documentation. The failure to continuously slope this line would allow air to hold up in the sensing lines and potentially prevent adequate venting. The inspectors presented this observation to the licensee who agreed that the discrepancy would result in the failures they experienced. The licensee determined the sensing line was most likely damaged due to human error during the fuel oil storage tank maintenance activities that involved work in the same area as the sensing line.

Analysis. The deficiency associated with this finding was the failure to maintain design control of the Train B emergency diesel fuel oil storage tank level instrument sensing line resulting in level indication error. This finding was greater than minor because it affected the mitigating systems cornerstone objective of ensuring the capability of emergency power to respond to initiating events to prevent undesirable consequences. Since the finding represented an actual loss of safety function, for a single train, for greater than its Technical Specification-allowed outage time, the finding was analyzed using Phase 2 of the Significant Determination Process. The finding was of very low safety significance because the licensee staff would have sufficient time to order replacement fuel; procedures existed to order replacement fuel; and training was conducted on the existing procedures under conditions similar to the initiating event assumed.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” states, in part, that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications and that deviations from such standards are controlled. The failure to maintain design control of the Train B emergency diesel fuel oil storage tank level instrument sensing line, resulting in level indication error, is a violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” Because the violation was of very low safety significance and has been entered into the licensee’s corrective action program as CR-W3-2005-02869, this violation is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy (NCV 0500382/2005003-01, Inadequate Design Control of the Train B Emergency Diesel Fuel Oil Storage Tank Level Instrument Sensing Line).

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Administrative Procedure ME-001-012, “Temporary Power from Temporary Diesel for 3A2 and 3B2 4kV Buses (Modes 1-6).” The inspectors reviewed the safety screening, design documents, UFSAR, and applicable Technical Specifications to determine that the temporary modification was consistent with the modification documents, drawings, and procedures. The inspectors walked down accessible portions of the affected equipment. The inspectors reviewed the adequacy of postinstallation tests and test results to confirm that the actual impact of the temporary modification on the permanent system and interfacing systems was adequately verified.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors inspected one emergency drill. On June 21, 2005, the inspectors reviewed the drill scenario and observed activities in the simulated control room. The drill scenario simulated instrument failures, loss of a charging pump, inadvertent main steam isolation valve closure, loss of offsite power, and a main steam line break in containment. The inspectors evaluated performance by focusing on the risk significant activities of emergency classification, notification, and protective action recommendations. In addition, the inspectors reviewed the drill critiques and the resolution of identified performance deficiencies.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiological Significant Areas (71121.01)

a. Inspection Scope

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

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

- Controls (surveys, posting, and barricades) of radiation, high radiation, locked high radiation, and areas with the potential for airborne radioactivity in the reactor containment building and reactor auxiliary building

- Radiation work permits, procedures, engineering controls, and air sampler locations

- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms.

- Barrier integrity and performance of engineering controls in five areas with the potential for airborne radioactivity

- Physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools.

- Self-assessments and audits related to the access control program since the last inspection

- Corrective action documents related to access controls

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

- Radiation work permit briefings and worker instructions

- Adequacy of radiological controls such as, required surveys, radiation protection job coverage, and contamination controls during job performance

- Dosimetry placement in high radiation work areas with significant dose rate gradients

- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

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

- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem CEDE

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

The inspector completed 21 of the required 21 samples.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

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

- Eight outage or on-line maintenance work activities scheduled during the inspection period and associated work activity exposure estimates which were likely to result in the highest personnel collective exposures.

- Site specific trends in collective exposures, plant historical data, and source-term measurements

- Site specific ALARA procedures

- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements

- Intended versus actual work activity doses and the reasons for any inconsistencies

- Assumptions and basis for the current annual collective exposure estimate, the methodology for estimating work activity exposures, the intended dose outcome, and the accuracy of dose rate and man-hour estimates

- Method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered
- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding
- Workers use of the low dose waiting areas
- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas
- Self-assessments, audits, and special reports related to the ALARA program since the last inspection

The inspector completed 7 of the required 15 samples and 4 of the optional samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators listed below for the period from September 2004 through March 2005. 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 3, 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 the licensee's Technical Specifications), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02) for the period of September 2004 through March 2005. Additional records 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.

Public Radiation Safety Cornerstone

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

  Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to the NRC for the period September 2004 through March 2005. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Annual Sample Review

a. Inspection Scope

The inspectors assessed implementation of Entergy's corrective action process involving multiple hydraulic actuator failures of shutdown cooling suction inboard containment isolation Valve SI-405B. The most recent failure occurred on April 17, 2005, when safety injection valve SI-405B did not fully open. This action prevented Entergy from placing shutdown cooling Train B in service for approximately 5 hours. The history of Valve SI-405B shows that the valve has continually failed to meet an acceptable standard of reliability. Although the shutdown cooling system has been in 10 CFR 50.65 (A)(1) status since March 2002, SI-405 B has failed to stroke open within its IST time limit in October 2003 and failed to open for shutdown cooling service in April 2005. A review of condition reports associated with SI-405B shows that the majority of conditions required an apparent cause and a fix of the problem. The inspectors noted there was limited information to determine the true cause(s) of the failures to operate based on the licensee's apparent cause evaluations. The inspectors did note, following the last failure, that Entergy performed a more thorough investigation into the overall reliability of this valve. This action has resulted in the licensee installing improved pressure and thermal relief valves for the actuator and implementation of more stringent test acceptance criteria to prevent future hydraulic actuator failures. The licensee is also evaluating the need for additional design changes to further improve its reliability.

b. Findings and Observations

No findings of significance were identified.

.2 Semiannual Trend Review

a. Inspection Scope

On June 24, 2005, the inspectors completed the semiannual review of Entergy's identified trends for evidence that other significant safety issues may exist. The inspectors' review focused on repetitive equipment issues, but also considered the results of screening the corrective action program, self-assessment reports, control room logs, quality assurance audits, and department self-assessments to determine if additional adverse trends existed. The inspectors compared and contrasted their results with the results contained in Entergy's latest quarterly trend reports. For those areas where trends were documented in the corrective action program, the inspectors verified that Entergy had corrective actions planned or in place to address the trend. The inspectors also evaluated the corrective actions against Entergy's procedural requirements of Procedure LI-102, "Corrective Action Program." The inspectors' review nominally considered the 6-month period of January through July 2005.

b. Findings and Observations

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

.3 The inspectors reviewed selected inservice inspection related condition reports issued during the current and past refueling outages. The review served to verify that the corrective action process was being correctly utilized to identify conditions adverse to quality and that those conditions were being adequately evaluated, corrected, and trended. The inspectors determined that the threshold for initiating condition reports was low, thereby, capturing any deficiencies identified in the inservice inspection program. The inspectors also concluded that corrective actions were being appropriately addressed. No findings of significance were identified.

.4 Section 2OS1 evaluated the effectiveness of the licensee's problem identification and resolution processes regarding access controls to radiologically significant areas and radiation worker practices. The inspector reviewed corrective action documents for root cause/apparent cause analysis against the licensee's problem identification and resolution process. No findings of significance were identified.

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

4OA3 Event Followup (71153)

a. Inspection Scope

On April 20, 2005, while the plant was performing reactor coolant system drain down activities for Refueling Outage 13, a partial vacuum condition was inadvertently induced in the reactor coolant system due to operator errors in establishing appropriate pressurizer and reactor head vent paths. This resulted in coolant boiling in the shutdown cooling suction line causing shutdown cooling pumps Trains A and B to experience motor amps and coolant flow oscillations. The inspectors assessed plant response to the transient with the review of control room logs, system process parameter trends, and interviews of control room operators and engineering personnel. The inspectors conducted control room observations during the recovery actions to verify proper cooling was maintained for the reactor fuel. Additionally, details surrounding the event were assessed and communicated to NRC management resulting in a determination that a special inspection was warranted.

b. Findings

Findings and observations related to this event were evaluated by an NRC special inspection and will be documented in NRC Inspection Report 05000382/2005-010.

4OA5 Other Activities

.1 TI 2515/161 - Transportation of Reactor Control Rod Drives in Type A Packages

a. Inspection Scope

This area was inspected to verify that the licensee's radioactive material transportation program complies with specific requirements of 10 CFR Parts 20, 71, and Department of Transportation regulations contained in 49 CFR Part 173. The inspector interviewed licensee personnel and determined that the licensee had undergone refueling/defueling activities between January 1, 2002, and present, but it had not shipped irradiated control rod drives in Department of Transportation Specification 7A Type A packages.

b. Findings

No findings of significance were identified.

2. Temporary Instruction 2515/163, "Operational Readiness of Offsite Power"

a. Inspection Scope

The inspectors collected data pursuant to TI 2515/163, "Operational Readiness of Offsite Power." The inspectors reviewed the licensee's procedures related to General Design Criteria 17, "Electric Power Systems;" 10 CFR 50.63, "Loss of All Alternating Current Power;" 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants;" and the Technical Specifications for the offsite power system. The data was provided to the Office of Nuclear Reactor Regulation for further review. Documents reviewed for this TI are listed in the attachment.

b. Findings

No findings of significance were identified.

.3 Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (PWRs) (TI 2515/160)

a. Inspection Scope

The objective of TI 2515/160, "Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors," was to support the NRC review of licensees' activities for inspecting pressurizer penetrations and steam space piping connections made from Alloy 82/182/600 materials, and to determine whether the inspections of these components are implemented in accordance with the licensee responses to Bulletin 2004-01, "Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at PWR." The purpose of this Bulletin was to: 1) advise PWR licensees that current methods of inspecting Alloy 82/182/600 materials used in the fabrication of pressurizer penetrations may require additional measures to detect and adequately characterize flaws due to primary water stress-corrosion cracking (PWSCC); (2) request PWR addressees provide the NRC with the pressurizer penetrations connections material fabrication information; and (3) request PWR licensees to provide the NRC inspection information to ensure that degradation of Alloy 82/182/600 pressurizer penetrations connections will be identified, adequately characterized, and repaired.

In response to Bulletin 2004-01, the licensee committed to perform a bare metal visual inspection of 100 percent of the Alloy 82/182/600 pressurizer penetrations using a VT-2 qualified examiner. On April 19, 2005, the inspectors observed the licensee performing the VT-2 inspection and performed a review, in accordance with a TI 2515/160, of the licensee's controls and personnel used for pressurizer penetration nozzles examinations met the licensee commitments contained in Bulletin 2004-01. The results of the inspectors' review per TI 2515/160 are listed below.

b. Observations

Summary: Based upon a bare metal visual examination of the pressurizer, the licensee identified small amounts of boric acid on pressurizer heater nozzle Sleeves C4 and D2 indicating pressure boundary leakage from the pressurizer. Eddy current examination confirmed that Nozzle C4 contained axial oriented cracks but failed to identify discernable defects associated with Nozzle D2. All existing pressurizer Alloy 600 nozzles were replaced with Alloy 690 nozzles during refueling outage 13 to preclude recurrence of primary water stress corrosion cracking.

Evaluation of Inspection Requirements

In accordance with the requirements of TI 2515/160, the inspectors have answered the following questions:

1. For each of the examination methods used during the outage, was the examination performed by qualified and knowledgeable personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

Yes. The licensee conducted a direct visual examination of the bare metal surface of all the pressurizer nozzles with a qualified examiner certified to Level III as a VT-2 examiner in accordance with procedure CEP-NDE-0112, “Certification of Visual Testing (VT) Personnel.” This qualification and certification procedure referenced the industry standards SNT-TC-1A, “Personnel Qualification and Certification in Nondestructive Testing,” and ANSI/ANST CP-189, “Standard for Qualification and Certification of Nondestructive Testing Personnel.”

2. For each of the examination methods used during the outage, was the examination performed in accordance with demonstrated procedures?

Yes. A 360 degree bare metal visual inspection was performed on all pressurizer penetrations in accordance with Entergy Engineering Procedure NOECP-107, “Administrative Control of Boric Acid Corrosion Control Program,” Change 2. The inspectors observed the licensee inspector performing the bare metal inspection of the pressurizer heater nozzles using qualified flashlights for VT inspection. Photographs were also taken to document inspection results. The licensee did demonstrate that adequate visual resolution was commensurate with an ASME Code VT-2 type inspection.

3. Able to identify, disposition, and resolve deficiencies?

Yes. The inspectors concluded that the licensee’s direct visual examinations were capable of detecting leakage from cracked pressurizer nozzle penetrations. This conclusion was based upon the inspectors direct observation of pressurizer penetration locations which were free of debris or deposits that could mask evidence of leakage in the areas examined.

4. Capable of identifying the leakage in pressurizer penetration nozzle or steam space piping components, as discussed in NRC Bulletin 2004-01?

Yes. The inspectors’ basis is discussed in question 3 above.

5. What was the physical condition of the penetration nozzle and steam space piping components in the pressurizer system (e.g., debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?

The lower pressurizer penetrations included 30 heater sleeve penetrations, sample line and surge line nozzles. The inspectors observed that all insulation was removed from the bottom of the pressurizer for the bare metal visual examination and performed a direct visual inspection for these pressurizer penetrations. Based on this examination, the area examined was clean and free of debris or deposits or other obstructions which could mask evidence of leakage. The inspector did not directly observe the pressurizer top head or side mounted instrumentation but noted that the licensee acceptance criteria for visual inspection for the top head and side mounted instruments were the same as for the Pressurizer bottom head nozzles.

6. How was the visual inspection conducted (e.g., with video camera or direct visual by the examination personnel)?

The licensee conducted a direct bare metal visual examination of these pressurizer penetrations. Photographs were taken of the nozzles that contained boric acid residue

7. How complete was the coverage (e.g., 360 degrees around the circumference of all the nozzles)?

The licensee performed a bare metal inspection 360 degrees around the circumference of each pressurizer penetration nozzle.

8. Could small boron deposits, as described in the Bulletin 2004-01, be identified and characterized?

Yes. The inspectors determined through direct observation of the licensee's efforts that the licensee staff were capable of detecting pressurizer nozzle leakage. The inspectors noted that identification of the boric acid deposits was left up to the judgment and training of the licensee's VT-2 qualified examiner but the licensee relied on the corrective action system process to make decisions on how to characterize deposits.

9. What material deficiencies (i.e., cracks, corrosion, etc.) were identified that required repair?

The licensee initially determined that pressurizer heater Sleeves C4 and D2 showed evidence of primary stress corrosion cracking. Subsequent eddy current examination determined that D2 contained no discernable detect. Heater Sleeve C4 contained axial cracks per eddy current examination. All existing pressurizer Alloy 600 nozzles were replaced with Alloy 690 nozzles during Refueling Outage 13 to preclude recurrence of primary water stress corrosion cracking. See 1R08 Inservice Inspection Activities for a volumetric inspection assessment of the newly installed Alloy 690 nozzles.

10. What, if any, impediments to effective examinations, for each of the applied methods, were identified (e.g., centering rings, insulation, thermal sleeves, instrumentation, and nozzle distortion)?

The licensee did not identify any impediments to an effective examination. All of the insulation had been removed around the nozzles to allow a direct visual examination of the bare metal for 360 degrees around the circumference of each penetration nozzle.

11. If volumetric or surface examination techniques were used for the augmented inspection examinations, what process did the licensee use to evaluate and dispose any indications that may have been detected as a result of the examinations?

Waterford-3 performed eddy current examination of the pressurizer heater Sleeves C4 and D2 during the RF13 outage. The examinations were performed with equipment and techniques equivalent to those used at the Palo Verde Nuclear Generating Station (PVNGS). The examinations were performed using a 3-coil probe with a 0.115" pancake coil, a mid-range +Point coil and a mag-biased +Point coil this probe is the same probe used at PVNS. The frequencies used in the examination were 400 kHz, 260 kHz, 100 kHz and 50 kHz. Data was acquired at 300 RPM at a pull speed of 0.15 inches per second. The data was acquired and analyzed on a Zetec MIZ-27 instrument. Calibration for the examination was performed on calibration standard Z-12972 the calibration standard contained ID axial and circumferential notches. This calibration standard was borrowed from the PVNGS for this examination. Data analysis was performed by a Level IIA qualified data analyst provided by Westinghouse Electric Co. and reviewed by a Level III qualified data analyst.

12. Did the licensee perform appropriate follow-on examinations for indications of boric acid leaks from pressure-retaining components in the pressurizer system?

Yes. All pressurizer nozzles that the licensee identified with boric acid deposit were evaluated per the procedure detailed in question 11.

c. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

.1 On April 29, 2005, the inspector presented the inspection results to Mr. J. Venable, Vice President, Operations, and other members of his staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

.2 The inspectors presented the results of the access controls inspection and the inservice inspection effort to Mr. J. Venable, Vice President, Operations and other members of licensee staff on May 6, 2005. Licensee management acknowledged the inspection findings.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. None were identified.

.3 On May 11, 2005, the inspector discussed the inspection findings with Mr. A. B. Pilutti, Radiation Protection Manager. The inspector verified that no proprietary information was provided during the inspection.

.4 The resident inspectors presented the inspection results to Mr. K. Walsh, General Plant Manager, and other members of licensee management at the conclusion of the inspection on July 6, 2005. The licensee acknowledged the findings presented.

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

4OA7 Licensee Identified Violations

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

.1 EDG B Open Field Flash Breaker

Technical Specification 6.8.1 requires written procedures be established, implemented, and maintained covering the activities contained in Appendix A of Regulatory Guide 1.33, Revision 2. Regulatory Guide 1.33, Appendix A, requires procedures for operation of emergency power sources, such as operation procedure OP-009-002, “Emergency Diesel Generator. OP-009-002 requires that the breaker for emergency diesel generator Train B field flash source (EG EBKR-15) be closed. Contrary to the above, on April 9, 2005, a Nuclear Auxiliary Operator discovered that the breaker for emergency diesel generator Train B field flash source (EG EBKR-15) was open. The licensee determined that the breaker was likely left open following maintenance on EDG B on April 4, 2005. This was identified in Entergy’s corrective action process as Condition Report CR-WF3-2005-1259. The finding was determined to be of very low safety significance based on the duration the breaker was open and the ability of operator actions to identify the condition and close the breaker using procedural guidance.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Anders, Superintendent, Plant Security
D. Boan, Health Physics Specialist, Radiation Protection
M. Bratton, Nondestructive Examination Level III
J. Brawley, Specialist, Health Physics
W. Brice, Specialist, Licensing
L. Dauzat, Supervisor, Radiation Protection Operations
C. Fugate, Assistant Manager, Operations (Shift)
T. Gaudet, Director, Planning and Scheduling
T. Mitchell, Director, Engineering
R. Murillo, Acting Manager, Licensing
D. Newman, Supervisor, Radiation Protection Instruments
R. O'Quinn, Code Programs and Senior Engineer, Steam Generators
R. Osborne, Manager, Programs and Components
G. Payne, Senior Project Manager, Engineering Projects
A. Harris, Acting Director, Nuclear Safety Assurance
C. Pickering, Code Programs Engineer
B. Pilutti, Manager, Radiation Protection
R. Redmond, Welding Engineer
G. Scott, Licensing Engineer
D. Stevens, Specialist, Health Physics
J. Venable, Vice President, Operations
K. Walsh, General Manager, Plant Operations

Other

Z. Cordero, Authorized Nuclear Inservice Inspector, ABS Consulting, Inc.

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

0500382/2005003-01	NCV	Inadequate Design Control of the Train B Emergency Diesel Fuel Oil Storage Tank Level Instrument Sensing Line

Closed

0500382/2005003-01	NCV	Inadequate Design Control of the Train B Emergency Diesel Fuel Oil Storage Tank Level Instrument Sensing Line

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedure

NUMBER	TITLE	REVISION
Operating Procedure OP-901-521	Severe Weather and Flooding	3

Condition Reports

CR-WF3-2005-01247

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
W3F1-97-0132	Waterford 3 Tornado Missile Protection	
W3-DBD-003	Emergency Feedwater System	2-7
W3-DBD-041	Safety Related HVAC	2-4

Section 1R04: Partial System Walkdown

Condition Reports

CR-WF3-2005-1259

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
Regulatory Guide 1.9	Selection, Design, Qualification and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants	0
W3-DBD-002	Emergency Diesel Generator & Automatic Load Sequencer	3

Work Orders

28112 52983866

Section 1R05: Fire Protection

Procedure

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

Section 1R07: Heat Sink Performance

Procedures

NUMBER	TITLE	REVISION
Special Test Procedure STP-01174749	EDG Heat Exchanger Performance Test	0

Miscellaneous Documents

NUMBER	TITLE	REVISION
W3-DBD-002	Emergency Diesel Generator & Automatic Load Sequencer	3
EC-M97-006	CCW Makeup Requirements	A
System Description	SD-EDG	7

Section 1R08: Inservice Inspection Activities (711111.08)

Procedures

Number	Title	Revision
WDI-ET-004	Intraspect Eddy-Current Analysis Guidelines	8
WDI-SSP-1002	RVH Penetration Inspection Tool Operation for ANO 2 and Waterford 3 - ROSA	1
MM-001-053	Control of Welding Filler Metal	11
QAP 9.6	Liquid Penetrant Inspection Procedure	10
QAP 9.20	Remote Liquid Penetrant Examination For Pressurizer Nozzle Mid-Wall Repair Welds	0
SI-UT-110	Ultrasonic Examination of Pressurizer Heater Sleeve Mid-Wall Repair Weld	1 and FMR001-FMR004
WPS 03-43-T-802 and Supporting PQR 03-43-T-803	Automatic Gas Tungsten Welding Procedure	2
UNT-006-031	Identification and Evaluation of Boric Acid Leakage	0
UNT-007-027	Boric Acid Walkdown	0
NOECP-107	Administrative Control of Boric Acid Corrosion Control Program	0
SI-NDE-08	Certification and Qualification of NDE Personnel Performing Examinations to ASME Section XI	0
USI SP-RT-1, App. 1-R	US Inspection Services Radiography Procedure	4
GQP 9.7	Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding	10
PI-900498-40E	Operational Qualification: Welding Power Supply, Dimetrics Goldtrack II System and PCI Block Weld Head	0
WPS 8MN-GTAW and PQRs 046R3, 062R3, and 600R2	Welding Procedure Specification	9
PI-900498-01	Replacement of Valve SI-407A	1
CEP-NDE-0423	Ultrasonic Examination of Austenitic Piping Welds (ASME Section XI)	0
SSPD	Site Specific Performance Demonstrations, Loose part training testing and training data.	6

ACTS WTR-01-05	Acquisition Technique Sheet Tubesheet Loose Parts, ETSS 96005.2, 96004.1	19
ACTS WTR-12-05	Ghent G3/G4 Mag Bias, ETSS 20406.1, 20407.1, 20507.1	0
ANTS WTR-A-05	Analysis Technique Sheet, Mix for eggcrates, ETSS 96008.1	6

Drawings

74170-101-003, Closure Hear, Penetrations - Final Machining, Revision 0
E-ESP-250-002, Tube Sheet Pattern, Steam Generator #1, Cold Leg
E-ESP-25-001, Tube Sheet Pattern, Steam Generator #1 Holt Leg

Welding Material Certifications (Certified Material Test Reports) For:

E 7018, 1/8", Heat 125184
E 7018, 3/32", Heat 12324
E308L-16, 3/32", Heat PP599
E308-16, 1/8", Heat 0N2A-2B
E308-16, 3/32", X39148
E309L-16, 1/8", Lot No. 2A5E-5A
E309L-16, 3/32", WPO 46572
E309-16, 1/8", WW092

Nondestructive Examination Reports

Radiography Report 900498-01RT on Field Weld 2A, Safety Injection Valve 407A
Radiography Report 900498-02RT on Field Weld 3A, Safety Injection Valve 407A
Liquid Penetrant Report 900498-05PT on Field Weld 3A, Safety Injection Valve 407A
Liquid Penetrant Report 900498-06PT on Field Weld 2A, Safety Injection Valve 407A

Liquid Penetrant Material Certifications For:

Spotcheck Developer, Batch 04J05K and 04C15K
Spotcheck Cleaner, Batch 04L12R and 04K02K
Spotcheck Penetrant, Batch 04F10R and 01M07K

Work Orders

WO 00040579-48
MAI 425045
425996
17184

Travelers

WSI 32024-01, “Work Traveler for Pressurizer Heater Sleeve Half Nozzle Repair with Mid-wall Sleeve Weld, Heater G3,” dated March 30, 2005

WSI 32024-01, “Work Traveler for Pressurizer Heater Sleeve Half Nozzle Repair with Mid-wall Sleeve Weld, Heater A4,” dated March 30, 2005

Miscellaneous Documents

CEP-ISI-001, “W3 Inservice Inspection Program,” Revision 9

Engineering Evaluation ER-W3-2003-0624-005, “Evaluate Corrosion on the Reactor Head Flange

ENS-DC-319, Entergy Nuclear South Boric Acid Corrosion Control Program (BACCP),” Revision 1

Request for Relief RR W3-R&R-003, dated March 31, 2005

Welder Qualification and Expiration Report, dated March 1, 2005

Filler Metal Request Form

Filler Metal Control Form

Welder Qualification Records for welders from PCI Energy Services (performed welding on Valve SI-407A)

Nondestructive Examination Personnel Qualification Records

Nondestructive Examination Procedure Qualification Demonstration Records

Technical Specification 3/4.4.4, “Steam Generators”

EPRI “Pressurized Water Reactor Steam Generator Examination Guidelines,” Revision 6

NRC Generic Letter 2004-01, Requirements for Steam Generator Tube Inspection”

PCI Qualification Certification for Welding Power Source and Weld Head Qualification, dated April 29, 2005

Linearity System Calibration Certification for UT Sonic-136 System, dated February 16, 2005

NRC Bulletin 2004-01, “Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors

NRC Information Notice 2004-11, “Cracking in Pressurizer Safety and Relief Nozzles and in Surge Line Nozzle”

NRC Information Notice 2004-08, “Reactor Coolant Pressure Boundary Leakage Attributable to

Propagation of Cracking in Reactor Vessel Nozzle Welds”

10 CFR 50.59 Review of Engineering Request ER-W3-2004-0122-000, which dealt with weld repair of 30 pressurizer heater sleeves, Revision 0

Condition Reports:

CR-WF3-2003-3130	CR-WF3-2005-1331	CR-WF3-2005-1413	CR-WF3-2005-1622
CR-WF3-2003-3512	CR-WF3-2005-1360	CR-WF3-2005-1449	CR-WF3-2005-1752
CR-WF3-2004-1354	CR-WF3-2005-1361	CR-WF3-2005-1516	CR-WF3-2005-1918
CR-WF3-2004-2577	CR-WF3-2005-1405	CR-WF3-2005-1528	CR-WF3-2005-1977
CR-WF3-2004-3924	CR-WF3-2005-1406	CR-WF3-2005-1540	CR-WF3-2005-01762
CR-WF3-2005-1305	CR-WF3-2005-1408	CR-WF3-2005-1587	

Section 1R15: Operability Evaluations

Procedures:

NUMBER	TITLE	REVISION
CEP-IST-1	IST Bases Document	3
OP-902-009	Standard Appendices	1.2

Condition Reports

CR-WF3-2005-1279	CR-WF3-2005-1172
CR-WF3-2003-1401	

Section 1R22: Surveillance Testing

Procedure

NUMBER	TITLE	REVISIONS
Calibration Procedure MI-005-205	Calibration/Functional Test of Level Instruments	8
UNT-006-031	Identification and Evaluation of Boric Acid Leakage	0
OP-903-024	Reactor Coolant System Water Inventory Balance	13
OI-040-000	Reactor Coolant System Leakage Monitoring	0
CEP-IST-1	IST Bases Document	3

Surveillance Procedure OP-903-068	Emergency Diesel Generator and Subgroup Relay Operability Verification	13

Work Orders

67776-01, 48615, 48593, 48598, 48567, 48617, 48603, 48610, 48609

Condition Reports

CR-WF3-2005-02869, CR-WF3-2005-02749, CR-WF3-2005-02750, CR-WF3-2005-02751, CR-WF3-2005-02752, CR-WF3-2005-02753, CR-WF3-2005-02754, CR-WF3-2005-02755, CR-WF3-2005-02756, CR-WF3-2005-02163, CR-WF3-2005-0913, CR-WF3-2004-03788, CR-WF3-2005-02979, CR-WF3-2005-02982

Section 2OS1: Access Controls to Radiologically Significant Areas (71121.01)

Audits and Self-Assessments

LO-WLO-2005-0005-01	ALARA Planning and Controls and Access to Radiologically Significant Areas
QA-14-2005-WF3-1	Radiation Protection

Corrective Action Documents

2005-1074, 2005-0113, 2005-1278, 2005-1298, 2005-1879, 2005-1887, 2005-1893, 2005-1931, and 2005-2011

Pre-Job ALARA Review Packages

2005-0602	RF-13 Alloy 600 Pressurizer Repair Project
2005-0617	Radwaste Activities in Containment during RF-13

Radiation Work Permit Packages

2005-0400	Radiography of Steam Generator Feedwater Piping and Safety Injection 407 “A” Valve, Revision 0
2005-0508	Inspect/Rework Reactor Coolant Pump Motors 1A, 1B, 2A, 2B and Change Out 2B Seal, Revision 0
2005-0511	Eddy Current Inspection and Tube Plugging Inside of Steam Generators Primary Side, Revision 0
2005-0702	Disassembly of Reactor Head and Associated Work Activities, Revision 1
2005-0708	In-Core Instrument Growth Project, Revision 0
2005-0717	Remove and Install Thimble Support Plate Stops, Revision 0

Procedures

EN-RP-104	Personnel Contamination Events, Revision 0
HP-001-107	High Radiation Areas Access Control, Revision 17
HP-001-243	Diving in Contaminated Waters Near Highly Radioactive Components, Revision 6
HP-002-201	Radiological Survey Techniques and Frequencies, Revision 17
RP-108	Radiation Protection Posting, Revision 2
RP-402	DOP Challenge Testing of HEPA Vacuums and Portable Ventilation Units, Revision 1
UNT-001-016	Radiation Protection, Revision 1
UNT-007-001	Control of Miscellaneous Material in the Spent Fuel Pool, Revision 4

Section 2OS2: ALARA Planning and Controls (71121.02)

Corrective Action Documents

CR-WF3-2004-3635, CR-WF3-2004-3682, CR-WF3-2004-3885, CR-WF3-2004-3878, CR-WF3-2004-3941, CR-WF3-2004-3974, CR-WF3-2004-4171, CR-WF3-2005-1692, CR-WF3-2005-1701, CR-WF3-2005-1677, CR-WF3-2005-1713, CR-WF3-2005-0993, CR-WF3-2005-0544, CR-WF3-2005-0690, CR-WF3-2005-0700, CR-WF3-2005-0756, CR-WF3-2005-0965, CR-WF3-2005-1233, CR-WF3-2005-1297, CR-WF3-2005-1299, CR-WF3-2005-0237, CR-WF3-2005-1706, CR-WF3-2005-1658, CR-WF3-2005-1670

Audits and Self-Assessments

LO-WLO-2005-0005-01	ALARA Planing and Controls and Access Control to Radiologically Significant Areas
QA-14-2005-WF--1	Radiation Protection

Radiation Work Permits

2005-0503	Alloy 600 Remove/Replace Nozzles on Both Hot Legs, Revision 0
2005-0610	Erect/Dismantle Scaffolding in the Reactor Containment Building, Revision 0
2005-0705	Reassembly the Reactor Head and all Associated Work Activities, Revision 0
2005-0707	Bullet Nose Removal/replacement, ICI Removal/Replacement, Revision 0
2005-0511	Perform Eddy Current work/Tube Plugging, Revision 0
2005-0610	Erect/Dismantle Scaffolding in the Reactor Containment Building, Revision 0
2005-0600	HP Surveys and Roving Job Coverage in the RCB and FHB, Revision 0
2005-0602	Alloy 600 Pressurizer Repair Project, Revision 0

Procedures

RP-105	Radiation Work Permits, Revision 5
RP-107	Radiation Protection Glossary, Revision 2
RP-108	Radiation Protection Posting, Revision 2
RP-109	Hot Spot Program, Revision 0
RP-110	ALARA Program, Revision 2
RP-103	Access Control, Revision2
HP-001-114	Control of Temporary Shielding, Revision 9
HP-001-243	Diving Operations in Contaminated Waters Near Highly Radioactive Components, Revision 6
UNT-01-016	Radiation Protection, Revision 1
UNT-05-027	Infrequently Performed Tests or Evolutions, Revision 2

Section 4OA1: Performance Indicator Verification (71151)

Procedure

EN-LI-114 Performance Indicator Process, Revision 0

Miscellaneous

NRC Performance Indicator Technique Sheets

Section 4OA2: Identification and Resolution of Problems

Condition Reports

CR-WF3-2005-1362	CR-WF3-2005-2127	CR-WF3-1998-0714
CR-WF3-2000-1455	CR-WF3-2002-0678	CR-WF3-2002-1056
CR-WF3-2003-2991	CR-WF3-2005-2837	CR-WF3-2005-2070
CR-WF3-2000-1347	CR-WF3-2005-3006	
CR-WF3-2002-0468	CR-WF3-2002-1523	

Miscellaneous Documents

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

Section 4OA5: Other

Condition Reports

CR-WF3-2005-1449
CR-WF3-2005-1752

Miscellaneous Documents

NUMBER	TITLE/SUBJECT	REVISION
NRC Bulletin 2004-01	Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors	0
UNT-006-031	Identification and Evaluation of Boric Acid Leakage	0
NOECP-107	Administrative Control of BACC Program	0
W3F1-2004-0058	Response to NRC Bulletin 2004-01 Regarding Inspection of Alloy 82/182/600 Materials Used in Pressurizer Penetrations and Steam Space Piping Connections	July 27, 2004
WOG-04-057	WOG CE Fleet Pressurizer Heater Sleeve Inspection Program	January 30, 2004

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
CFR	*Code of Federal Regulations*
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
NRC	Nuclear Regulatory Commission
PDR	Public Document Room
PVNGS	Palo Verde Nuclear Generating Station
PWR	Pressurized Water Reactors