ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-382
License No.:	NPF-38
Report No.:	50-382/97-11
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	May 18 through June 28, 1997
Inspectors:	L. A. Keller, Senior Resident Inspector W. F. Smith, Senior Resident Inspector, River Bend R. V. Azua, Project Engineer G. E. Werner, Project Engineer
Approved By:	P. H. Harrell, Chief, Project Branch D

Attachment: Supplemental Information

EXECUTIVE SUMMARY

Waterford Steam Electric Station, Unit 3
NRC Inspection Report 50-382/97-11

This routine, announced inspection included aspects of licensee event response, operations, maintenance, engineering, and plant support. The report covers a 6-week period of resident inspection.

Operations

- Observed operations activities including reactor coolant system (RCS) reduced inventory operations were generally well coordinated and consistent with safe operation of the facility (Section O1.1).

- A violation was identified regarding the failure to implement procedural requirements for a tagout which contributed to the spill of approximately 5000 gallons of radioactive water from the spent fuel pool (SFP) (Section O1.2).

- Control room operators demonstrated poor judgement in not promptly clearing an SFP high-level alarm (Section O1.2).

- The licensee root cause investigation into the SFP spill event was thorough and accurate (Section O1.2).

- Operators responded very effectively to a partial loss of offsite power while at midloop (Section O4.1).

Maintenance

- Weaknesses in the licensee's work control process resulted in a technician being provided a work package that required revision, despite the errors being previously identified (Section M1.2).

- Emergency diesel generator (EDG) integrated safeguards testing was well planned and executed (Section M1.3).

Engineering

- Engineering generally provided good technical support to operations and maintenance (Section E1.1).

- The proposed design changes to provide 13 air-operated containment isolation valves with the capability to be closed remotely and remain closed for 30 days after the design-basis accident were found to be adequate (Section E2.1).

- The procedure changes required as a result of the implementation of a modification were not being tracked to ensure completion prior to entry into Mode 4 (Section E2.1).

Plant Support

- Observed radiation protection activities were performed in accordance with procedures and were consistent with ALARA principles. The small amount of contaminated areas continued to be a strength (Section R1.1).

- Planning and preparation for the upcoming hurricane season were good (Section P1.1).

- The site drill conducted on June 11, 1997, provided good training for emergency response personnel. Plant management's willingness to conduct a full participation drill during the refueling outage demonstrated commitment to excellence in emergency preparedness (Section P1.1).

Report Details

Summary of Plant Status

Throughout this inspection period the plant was shut down for Refueling Outage (RFO) 8.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors performed frequent reviews of ongoing plant operations, control room board walkdowns, and plant tours. Observed activities were generally performed in a manner consistent with safe operation of the facility. The inspectors observed good operator performance during RCS reduced inventory operations and a partial loss of offsite power event. However, an operator procedural violation and failure to clear an annunciator contributed to an SFP spill event as discussed below.

O1.2 SFP Spill

a. Inspection Scope (93702, 71707)

The inspectors reviewed the circumstances involving the inadvertent overflow of the SFP on May 21, 1997.

b. Observations and Findings

At approximately 9 p.m. on May 20, the shift support center (SSC) was approached by a plant mechanic with Work Authorization 01159118, which involved the repair of a body-to-bonnet leak on Valve FS-405, an SFP purification system isolation valve. The SSC operator (a licensed reactor operator) prepared clearance tags for Valve FS-405 under an existing tagout, Tagout 97-0864. Tagout 97-0864 was issued previously to isolate drains from the refueling cavity during refueling. In preparing the clearance tags for Valve FS-405, the SSC operator failed to consider whether the refueling water storage pool (RWSP) purification pump might be running and did not inform the control room that additional tags were being added to an existing tagout. Additionally, the required independent review was not performed prior to issuing the tags to be hung in the field. The inadequate tagout preparation resulted in the failure to secure the RWSP purification pump prior to hanging the tags. This resulted in dead-heading the RWSP purification pump, which in turn resulted in higher than normal system pressure and subsequent leakage into the SFP via Isolation Valve FS-345.

At 10:45 p.m. on May 20, the SFP high-level alarm annunciated in the control room. In response to the alarm: (1) an operator locally verified SFP level was at the high-level mark of 44 feet, which represented a 1-inch rise in level from when it was checked 24 hours earlier, (2) SFP influent valves were verified closed (including FS-345), (3) chemistry sampled the SFP, which indicated no dilution of the SFP, and

(4) the auxiliary operator was instructed to periodically check the SFP level. The operators discussed the possible cause of the high-level alarm and speculated that, since the component cooling water (CCW) makeup pump (a source of makeup to the SFP) had been running, there could have been leakage past the CCW makeup pump isolation valve into the SFP. The crew discussed the need to lower the SFP level and clear the alarm; however, this was given a low priority due to the false assumption that the SFP level rise had stopped (or was very slow) and due to ongoing outage evolutions (i.e., EDG run and high-pressure safety injection system venting). At this point, the control room was blind to any further SFP increase since there was no SFP level indication in the control room and the high-level alarm remained energized.

At 2:41 a.m. on May 21, the control room received reports that the SFP was overflowing into the spent fuel cask decon pit and from there draining into the fuel handling building (FHB) railroad bay. At 2:46 a.m., RWSP purification was secured, which stopped the ingress of water into the SFP. Approximately 5000 gallons of radioactive water overflowed from the SFP into the FHB; of this, approximately 2500 gallons were contained in the FHB railroad bay. The licensee estimated that between 230 and 1850 gallons escaped outside the FHB through the railroad bay doors where it spread out over a large area of asphalt and gravel within the protected area and into the storm drain system. The remainder of the spilled fluid was captured in the reactor auxiliary building sump and waste systems.

Immediate corrective actions included pumping excess SFP water to the RWSP, draining the cask decon pit, isolating the spill area to prevent spread of contamination, and taking soil and liquid samples to determine if any reportable releases to the environment occurred. Sample results indicated that the effluent concentration limits of 10 CFR Part 20 were not exceeded. Condition Report (CR) 97-1284 was initiated and an event review team was convened on May 21 to investigate the event. Throughout the next several weeks, extensive cleanup was conducted, which included removal of contaminated gravel, soil, and asphalt and flushing of the storm drain system. The inspectors observed portions of the cleanup activities and concluded that the spill was adequately contained and that no 10 CFR Part 20 limits were exceeded.

The licensee's event review team concluded that the spill was the result of a combination of tagging and communication errors, which resulted in dead-heading the RWSP purification pump combined with an SFP purification isolation valve that leaked. Contributing to the event was the failure to promptly clear the SFP high-level alarm due to other ongoing shift evolutions. The event review team's investigation revealed that the travel stop nuts on SFP Isolation Valve FS-345 were incorrectly positioned 1/8 inch lower than required by the valve technical manual during a maintenance activity in May 1992. This resulted in the valve not fully blocking flow when it indicated shut. The inspectors reviewed the event review team report and concluded that it was thorough and accurate.

Administrative Procedure UNT-005-003, "Clearance Request, Approval, And Release," Revision 14, Section 5.3, "Preparation of the Clearance Form," requires, in part, that if tags are added or deleted to a clearance, the individual making the changes shall ensure that the clearance affords the same level of protection or better than the original clearance, the clearance shall receive an independent review by a licensed operator to verify that the boundaries chosen are adequate, and the clearance is forwarded to the shift supervisor/control room supervisor for review. The failure of the SSC operator to comply with the tagging procedure requirements is a violation of Technical Specification (TS) 6.8.1.a (50-382/9711-01).

c. Conclusions

The SFP spill was the result of an inadequate tagout due to procedure noncompliance, poor communications between the SSC and the control room, and an isolation valve not being fully closed. Control room operators demonstrated poor judgement in not promptly clearing the SFP high-level alarm. The spill was adequately contained and no 10 CFR Part 20 limits were exceeded. The licensee root cause investigation into the SFP spill event was thorough and accurate.

O4 Operator Knowledge and Performance

O4.1 Startup Transformer (SUT) Failure (93702, 71707)

On May 28, 1997, the plant was in Mode 5 with the RCS at midloop in support of a reactor coolant pump seal replacement. Both shutdown cooling trains were in service with the decay heat load evenly split between the two trains. At 9:01 a.m., an internal fault occurred in SUT B which resulted in the loss of Train B offsite power. EDG B started and energized the Train B safety loads per design. As a result of the event, SFP cooling and Shutdown Cooling Train B were temporarily lost, with only negligible temperature increases in the SFP and RCS prior to their being reenergized. The inspectors responded to the control room within minutes of the SUT failure. The inspectors observed very good command and control, formal communications, and good procedural compliance. At 9:40 a.m., the control room supervisor held a briefing on current status and priorities. The inspectors observed good decision making during the briefing, including the decision to stop noncritical work in the plant and restrict access to EDG A and other Train A electrical components.

At 11:30 p.m., on May 29, the Train B loads were reenergized from offsite power by backfeeding through the main transformer. EDG B was subsequently unloaded and secured after having supplied the Train B loads for over 38 hours without incident. Train B was energized via backfeed from the main transformer throughout the remainder of this inspection period. SUT B was shipped offsite for disassembly, inspection, and repair. A replacement transformer was subsequently purchased and delivered onsite. As of the end of the inspection period, the replacement transformer was undergoing receipt inspections in preparation for testing and eventual installation.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) Unresolved Item 50-382/9613-02: Failure to properly isolate CCW containment penetration.

During development of work procedures for Valve CC-807A, a containment fan cooler CCW inlet isolation valve, operators determined that they could not isolate the containment penetration since only a check valve existed between the penetration and the temporary chiller system. The inspectors were in the control room while this issue was being discussed and recalled that a check valve had been credited for containment penetration isolation during a similar maintenance activity performed from October 23-25, 1996, for Valve CC-808A, another containment fan cooler CCW inlet isolation valve, which isolated Containment Penetration 20. Subsequently, the licensee initiated CR 96-1726 to review and evaluate the circumstances related to isolating the containment penetration. The licensee determined that operators had not isolated all flow paths by use of a deactivated automatic valve, a manual valve, or a blind flange as specified in the actions for TS 3.6.3. Instead, operators had used a check valve as a containment isolation valve barrier for the penetration. This issue was identified as unresolved pending review of Task Interface Agreement 96TIA017, which requested that the Office of Nuclear Reactor Regulation (NRR) resolve the regulatory requirements associated with isolation and closure capability for the containment fan cooler isolation valves.

The inspectors reviewed the response to Task Interface Agreement 96TIA017 and determined that a violation of TS 3.6.3 occurred. As part of the corrective actions, the licensee discussed this issue with all operators and issued a letter reinforcing TS operability expectations. In addition, the licensee is planning to modify: (1) CCW prints to provide easier interpretation, and (2) the CCW system to provide positive train boundaries for the temporary chill water supply lines.

Although the inspectors questioned this issue before the CR was written, the licensee identified that it was inappropriate to credit the check valve, and the inspectors concluded that a CR would have been initiated absent the inspectors question. This licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. Specifically, the violation was identified by the licensee, was not willful, actions taken as a result of a previous violation should not have corrected this problem, and appropriate corrective actions were completed by the licensee (50-382/9711-02).

O8.2 (Closed) Violation 50-382/9603-03: Inoperable steam supply system for turbine-driven emergency feedwater (EFW) pump.

This item involved the failure to perform the required TS actions because operations incorrectly assumed only one admission valve was required for operability. Corrective actions included training of all operations personnel and a change to Procedure OP-100-014, "Technical Specification Compliance," which added specific guidance that both steam supplies for the turbine-driven EFW pump are required to be operable. This item was part of a generic concern regarding failure to enter appropriate TS limiting conditions for operations. The generic concern was dispositioned under Unresolved Item 50-382/9605-03, which was closed in NRC Inspection Report 50-382/96-13.

O8.3 (Closed) Violation 50-382/9613-03: Failure to enter appropriate TS for inoperable wet cooling tower (WCT) fans.

This item involved the failure of operators to recognize that curtains placed in front of the WCT rendered it inoperable. This violation was caused by engineering input that did not clearly communicate that a TS entry would be required and an inadequate review of the work package by operators. Corrective actions included removing the curtains, eliminating the use of engineering inputs to make operability determinations, and incorporating this incident into training for operators. This item was part of a generic concern regarding failure to enter appropriate TS limiting conditions for operations. The generic concern was dispositioned under Unresolved Item 50-382/9605-03, which was closed in NRC Inspection Report 50-382/96-13.

II. Maintenance

M1 Conduct of Maintenance (62707, 61726)

M1.1 General Comments

The inspectors observed all or portions of the following maintenance and surveillance activities:

- DCP-3623 Override of Safety Injection Actuation Signal (SIAS) and Containment Spray Actuation Signal (CSAS) for Containment Isolation Valves
- OP-903116 Train B Integrated EDG Engineering Safety Features Test

M1.2 Override of SIAS and CSAS for Containment Isolation Valves

a. Inspection Scope (62707)

The inspectors reviewed Design Change Package (DCP) 3623, "Override of SIAS and CSAS for Containment Isolation Valves," and observed selected maintenance activities associated with the DCP.

b. Observations and Findings

On May 23, the inspectors observed portions of the maintenance activities being performed under DCP 3623. The technician performing this activity stated that he had not progressed very far because he had discovered errors in the DCP. He then stated that when he notified design engineering of the errors, he was informed that another technician, who had been working on the DCP prior to him, had identified the same errors and that a Document Revision Notice (DRN I-9701799B) had already been issued to address the errors. The technician stated that he obtained a copy of this DRN from document control. The inspectors were concerned that the technician was given a work package that was in error and required revision, despite the errors being previously identified. The inspectors were informed that the original technician responsible for this activity had called in sick and, as a result, no turnover was provided when the responsibility for completion of this effort was reassigned. The inspectors noted that no log was maintained with the DCP that provided any insight to the fact that errors had been found in the DCP and that a DRN revision was being developed.

The licensee's Administrative Procedure MD-001-014, "Conduct of Maintenance," did not provide any guidance regarding turnovers. The licensee stated that, normally, an effort was made to keep the same technician working on a task through completion to minimize the potential for these types of errors. The licensee issued CR 97-1306 to address this problem. In addition, the maintenance department instituted the use of a job status log, which will be added to all work packages.

The inspectors questioned whether a process existed by which craft personnel were notified when a DRN was issued. The licensee indicated that it was the responsibility of the document control department to make such notifications. The inspectors reviewed the licensee's Site Procedure W5.201, "Document Control Systems," and Procedure UNT-004-009, "Handling and Use of Technical Documents," and found that, although a requirement existed for notification of controlled document holders regarding revisions that had been issued, no specific guidance on timeliness or how that contact was to be made was provided. The inspectors found that, although the document control department had a process for notification, which included a data base for recording such notifications, it was not proceduralized. The inspectors questioned if their records indicated that the holder of the controlled copy of DCP 3623 had been notified of the issuance of

DRN I-9701799B. They stated that their records did not indicate that such a notification was made. The licensee subsequently discovered 13 other examples where the document control program failed to notify document holders of revisions to their documents. In all cases, the end users were aware of the revisions and therefore no work errors had resulted due to the notification oversight.

c. Conclusions

The lack of any turnover or requirements for a turnover, specifically in the case of a complex work package, was considered a weakness in the licensee's maintenance program. The failure by the document control department to notify the holders of controlled documents regarding revisions to their work packages is a weakness in the licensee's document control program.

M1.3 Train B Integrated EDG Engineering Safety Features Test (61726)

On June 20, 1997, the inspectors monitored the licensee's activities in preparation for and performance of Procedure OP-903116. The licensee held a prejob brief in accordance with licensee requirements for infrequently performed activities. The briefing was found to be thorough, including a discussion on the steps necessary to back out of the surveillance test in the event of any problems. The surveillance coordinator was clearly identified and the responsibility for each person involved was clearly specified. During the performance of the surveillance, the licensee performance was found to be good with no problems noted.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Closed) Inspection Followup Item 50-382/9603-02: Review licensee's testing methodology.

This item involved the potential for preconditioning for timing the opening of the steam admission valves for the turbine-driven EFW pump. The licensee, by procedure, timed the valves on the second pump start rather than the first. The first start of the pump tested the turbine mechanical overspeed trip device, and the second start timed the opening of the steam admission valves. The licensee revised Procedure OP-903-046, "Emergency Feed Pump Operability Check," so that the steam admission valves would be tested in the as-found condition. The inspectors noted that the stroke times for these valves remained satisfactory and did not appreciably change as a result of the change in testing methodology and, therefore, previous testing had not masked any valve degradation. The inspectors concluded that the procedure enhancements were adequate to resolve the preconditioning concern.

III. Engineering

E1 Conduct of Engineering

E1.1 General Comments (37551)

In general, engineering provided good technical support to operations and maintenance. Daily reviews of CRs indicated engineering personnel had an appropriately low threshold for identifying problems.

E2 Engineering Support of Facilities and Equipment

E2.1 Correction of Air-Operated Containment Isolation Valve Deficiencies

a. Inspection Scope (37551)

The inspectors reviewed the design change (DC) initiated by the licensee to correct deficiencies in the original design of 13 containment isolation valves. These valves were designed and installed to fail open to support the associated system safety functions, but were not capable of being closed and maintained closed for 30 days, as required by the Standard Review Plan, Section 6.2.1, should the containment isolation function be needed concurrent with a loss of nonsafety-related instrument air. The scope of this inspection was to evaluate the adequacy of licensee actions to correct this problem prior to restart from RFO 8.

b. Observations and Findings

NRC Inspection Reports 50-382/96-24 and 97-13 addressed the following safety concerns:

- Air-operated Containment Spray Isolation Valves CS-125A(B) were designed to open on a CSAS and to fail open on a loss of nonsafety-related instrument air to the operator. However, Valves CS-125A(B) were also designated containment isolation valves, which were required to be capable of being closed manually from the control room and remain closed for 30 days following an accident. However, these valves were unable to be closed in the presence of a CSAS, nor were they capable of remaining closed for 30 days if instrument air was lost.

- Air-operated CCW to Containment Fan Cooler Isolation Valves CC-807A(B), -808A(B), -822A(B), and -823A(B) were in a circumstance similar to the above containment spray valves, except that the valves did not have remote manual switches in the control room. These valves automatically opened and closed in response to their respective containment fan cooler fan operation, but could not be closed in the presence of an SIAS.

The licensee had previously identified Containment Isolation Valves CC-641, -710, and -713, which were designed to fail open to ensure CCW cooling for the reactor coolant pump shaft seals and the control element drive mechanism fans; however, the three valves were not designed to remain closed for 30 days upon loss of instrument air. These valves automatically closed on a CSAS. Consideration of a supplemental source of instrument air was addressed in Licensee Event Report 50-382/92-015, when it was discovered that postaccident radiation levels in the vicinity of the manual operators for Valves CC-641 and -713 could be higher than postulated in the Updated Final Safety Analysis Report.

In addition, the licensee identified that outboard charging system Containment Isolation Valve CVC-209 was designed to fail open upon loss of instrument air and the valve had no accumulator to hold the valve closed. Further, the solenoid control valve supplying air to the actuator and the position indication were not Class 1E. For the short term, the licensee elected to correct this problem by designating remote reach-rod actuated Manual Isolation Valve CVC-208 as the containment isolation valve in place of Valve CVC-209. Valve CVC-208 was 3 feet upstream of Valve CVC-209 and could be operated from Switchgear Room B within 10 minutes from the time it was identified that the containment isolation function was needed for Valve CVC-209. The valves and piping were Seismic Category 1 and Safety Class 2. Additional barriers upstream of Valve CVC-208 included the charging pump discharge check valves, the positive displacement pump internal valves, and the charging pump suction check valve. The containment penetration isolated by Valve CVC-209 (Containment Penetration 27) is listed in Updated Final Safety Analysis Report Table 6.2-43 as not requiring a local leak rate test because it is connected to a closed Seismic Category 1, Safety Class 2, system. In a letter to the NRC dated May 6, 1997, the licensee committed to restore Valve CVC-209 to full compliance with NRC regulations during the next RFO by providing 30-day closure capability and upgrading the controls and indication to Class 1E in accordance with DC 3529. This allowed time for final project scoping, material lead time, and DCP development. The inspectors considered the licensee's short-term corrective action acceptable for startup from RFO 8.

The inspectors reviewed DC 3429, "Essential Air Supply to Fail Open Containment Isolation Valves," Revision 3. The documentation in DC 3429 provided a backup supply of safety-related instrument air so that, if there was a need to isolate any of the containment penetrations protected by any of the above 13 valves and there was a loss of nonsafety-related instrument air, the isolation could be accomplished for at least 30 days, and the operators could place the backup air supply in service from a low radiation area after a design-basis accident. The DC package contained all of the elements required by the licensee's administrative controls over design changes, including a safety evaluation pursuant to 10 CFR 50.59. The safety evaluation did not identify any unreviewed safety questions and had sufficient basis to support that conclusion.

Briefly, each of the 13 valves had a safety-related accumulator connected to the air actuator capable of closing each valve and holding it closed for 6-10 hours upon loss of nonsafety-related instrument air in accordance with the original design. DC 3429 consisted of installing and connecting a safety-related, Seismic Category 1, stainless steel air manifold to the accumulators such that the safety-related air supply would last for 30 days at the design maximum leakage rate. To maintain safety train separation and minimize tubing runs, four high-pressure stations were to be installed, each consisting of five refillable high-pressure air bottles that would be maintained at 2250-2500 psig. Each station had its own reducing valve, which reduced the pressure to approximately 98 psig. A relief valve was provided to protect the low-pressure side of the four subsystems, and the high-pressure air bottles were protected by the standard rupture disc. Twelve of the 13 accumulators for the valves listed above were located outside containment; however, the accumulator for Valve CC-710 was located inside containment. An existing service air penetration was utilized to provide a path for the supplemental air source.

The new air stations were to be normally isolated from the valve accumulators. The operators would place the air stations in service by opening two manual valves on each station located on the auxiliary building 21-foot elevation in an area of low postaccident radiation. The added containment isolation valve for the air supply to Valve-CC-710 was a solenoid valve that was controlled from Control Room Panel CP-8. With the existing accumulators installed and periodically tested, the operators would have several hours to get to the stations if nonsafety-related instrument air was lost during or after an accident.

Acceptance testing requirements specified in the package for DC 3429 were appropriate and comprehensive. Although the special test procedure was not fully written as of the time of this inspection, the licensee indicated that it was going to reflect the DC package requirements as a minimum.

During this inspection, the inspectors performed a field check of the installation and found that only about half of the tubing runs were completed and the air station installations were not started. However, the tubing and supports that were installed appeared to be of good quality and workmanship.

The inspectors reviewed DC 3523, "Override of SIAS & CSAS for Containment Isolation Valves," Revision 0. This design change was completed except for the acceptance testing and revision of affected procedures. DC 3523 installed key-lock switches on Auxiliary Panels 1 and 2 in the relay room located on elevation +35 feet. The switches were wired to allow manual remote closing of Containment Spray Isolation Valves CS-125A(B) and CCW to Containment Fan Cooler Isolation Valves CC-807A(B), -808A(B), -822A(B), and -823A(B) when there was a CSAS and SIAS present, respectively, and the containment isolation feature was needed. DC 3523 also installed annunciators in the control room designed to alarm when the override key-lock switches were out of the normal nonoverride position.

The inspectors performed a walkdown inspection of the DC 3523 installation in the plant. The key-lock switches were installed on Auxiliary Panels 1 and 2 with the correct labeling in accordance with the DC package. The inspectors also noted that the annunciator windows were installed in the control room with the correct wording to indicate the presence of an override signal.

The inspectors reviewed Special Test Procedure STP-99003523, "DC 3523 Acceptance Test," Revision 0, which was reviewed by the Plant Operations Review Committee and approved by the General Manager, Plant Operations. The procedure had sufficient scope to thoroughly test the installation for operability. The inspectors had a few minor comments, which were resolved with the procedure author.

The inspectors questioned how the licensee was managing the timely revision of procedures affected by DC 3429 and DC 3523. Of particular concern were several operating, abnormal operating, and annunciator response procedures. The inspectors found that the operations department did not have a listing of those procedures that were required to be revised and implemented in support of proceeding to Mode 4. The inspectors were subsequently informed that a list was generated identifying procedures that were considered Mode 4 constraints.

c. Conclusions

The licensee implemented appropriate design changes to the plant, thereby providing 13 identified air-operated containment isolation valves with the capability to be closed remotely and remain closed for at least 30 days after the design-basis accident. This inspection activity established the necessary confidence that the safety issue will be resolved upon satisfactory completion of the work and acceptance testing prior to placing the plant in Mode 4 and subsequent power operations.

The short-term compensatory actions addressing outboard charging system Containment Isolation Valve CVC-209 were appropriate to the circumstances and suitable for entry into Mode 4 and subsequent power operations until RFO 9, when long-term corrective actions are scheduled to be completed.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 General Comments (71750)

Routine tours of the radiological controlled area revealed that: (1) posting of areas was in accordance with requirements, (2) controlled access areas were properly locked, (3) personnel were wearing appropriate dosimetry and protective clothing, and (4) the small number of contaminated areas continued to be a strength.

The inspectors concluded that observed radiation protection activities were performed in accordance with procedures and were consistent with ALARA principles.

P1 Conduct of EP Activities

P1.1 Hurricane Season Preparations (71750)

The inspectors reviewed the licensee's preparations for the upcoming hurricane season. Planning and preparations appeared thorough. Some of the preparations included:

- walkdowns of dedicated hurricane storage areas and supplies,
- training on the use of satellite communications systems for duty emergency planners,
- All Hands Meetings conducted on May 20, 22, and 27, during which information packets were distributed,
- participation in a joint hurricane tabletop drill with the St. Charles Parish Emergency Preparedness Director,
- incorporated nuclear industry lessons learned from Hurricanes Bertha and Fran into plant procedures.

P1.2 June 11, 1997, Site Drill

The inspectors observed the site drill from the technical support center. The inspectors noted that the drill scenario was challenging. Overall drill performance was good. The inspectors noted some minor performance weaknesses, but these items were independently identified and critiqued by the drill evaluators. The inspectors concluded that licensee management's willingness to perform this challenging, full participation drill during their RFO demonstrated a commitment to excellence in the area of emergency preparedness.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on July 3, 1997. 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.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

C. M. Dugger, Vice-President, Operations
E. C. Ewing, Director Nuclear Safety & Regulatory Affairs
T. J. Gaudet, Manager, Licensing
J. G. Hoffpauir, Acting Manager, Maintenance
T. R. Leonard, General Manager, Plant Operations
D. C. Matheny, Manager, Operations
D. W. Vinci, Superintendent, System Engineering
A. J. Wrape, Director, Design Engineering

INSPECTION PROCEDURES USED

37551	Onsite Engineering
61726	Surveillance Observations
62707	Maintenance Observations
71707	Plant Operations
71750	Plant Support Activities
92901	Followup - Plant Operations
92902	Followup - Maintenance
93702	Event Response

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-382/9711-01	VIO	Failure to follow tagging procedure (Section O1.2.b)
50-382/9711-02	NCV	Failure to properly isolate CCW containment penetration (Section O8.1)

Closed

50-382/9613-02	URI	Failure to properly isolate CCW containment penetration (Section O8.1)

50-382/9711-02	NCV	Failure to properly isolate CCW containment penetration (Section O8.1)
50-382/9603-03	VIO	Inoperable steam supply system for turbine-driven EFW pump (Section O8.2)
50-382/9613-03	VIO	Failure to enter appropriate TS for inoperable WCT fans (Section O8.3)
50-382/9603-02	IFI	Review licensee's testing methodology (Section M8.1)

LIST OF ACRONYMS USED

ALARA	as low as reasonably achievable
CCW	component cooling water
CFR	Code of Federal Regulations
CR	condition report
CSAS	containment spray actuation signal
DC	design change
DCP	design change package
DRN	document revision notice
EFW	emergency feedwater
EDG	emergency diesel generator
FHB	fuel handling building
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
psig	pounds per square inch gauge
RCS	reactor coolant system
RFO	refueling outage
RWSP	refueling water storage pool
SFP	spent fuel pool
SIAS	safety injection actuation signal
SSC	shift support center
SUT	startup transformer

TS	Technical Specification
WCT	wet cooling tower