UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4005

October 31, 2008

Kevin T. Walsh, Vice President,
Operations
Waterford 3
Entergy Operations, Inc.
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000382/2008-004

Dear Mr. Walsh:

On September 16, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Waterford Steam Electric Station, Unit 3. The enclosed report documents the inspection results, which were discussed on September 30, 2008, with you and members of your staff.

The 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. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, two NRC-identified findings of very low safety significance were identified. The findings involved violations of NRC requirements. Additionally, one licensee-identified violation, which was determined to be of very low safety significance, is also listed in this report. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating these issues as noncited violations in accordance with Section VI. A. 1 of the NRC Enforcement Policy.

If you contest the subject or severity of any of the noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region IV, 611 Ryan Plaza, Suite 400, Arlington, TX 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office 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 and its enclosure 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), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Charles J. Paulk, Chief
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Division of Reactor Projects

Docket: 50-382
License: NPF-38

Enclosure: Inspection Report 05000382/2008004
w/Attachment: Supplemental Information

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SUNSI Review Completed: _CJP___ADAMS: ☑ Yes ☐ No Initials: __CJP__
☑ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☑ Non-Sensitive

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RIV:SRI:DRP/E	RI:DRP/E	SPE:DRP/E	C:DRS/EB1	C:DRS/OB
RVAzua	DHOverland	GDReplogle	RLBywater	RELantz
/RA/CP for	***/RA – E/***	***/RA – E/***	***/RA/***	***/RA/CP for***
10/31/08	10/29/08	10/30/08	10/29/08	10/29/08
C:DRS/EB2	C:DRS/PSB1	C:DRS/PSB2	C:DRP/E	
NFOkeefe	MPShannon	GWerner	CPaulk	
/RA/CFO for	***/RA/JL for***	***/RA/LR for***	***/RA/***	
10/29/08	10/29/08	10/29/08	10/31/08	

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U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-382
Licenses:	NPF-38
Report:	05000382/2008004
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, LA
Dates:	July 1 through September 16, 2008
Inspectors:	R. Azua, Senior Resident Inspector D. Overland, Resident Inspector G. Replogle, Senior Project Engineer R. Smith, Senior Resident Inspector W. Sifre, Senior Reactor Inspector S. Graves, Reactor Inspector B. Henderson, Reactor Inspector S. Makor, Reactor Inspector
Approved By:	Charles J. Paulk, Chief Project Branch E Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000382/2008-004; 07/01/2008 – 09/16/2008; Waterford Steam Electric Station, Unit 3; Operability Evaluations, Postmaintenance Testing.

This report covers a 3-month period of inspection by the resident and regional inspectors. Two Green noncited violations of very low safety significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 609, "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 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of Technical Specification 6.8.1.c (Procedures) for the failure to open the Train A low pressure safety injection pump suction valve prior to pump operation during a surveillance. The butterfly valve was installed 90 degrees out of position and was closed when operators believed it was open. After starting the pump, operators observed loud noises coming from the unit and secured it 8 minutes later. Pump operation without adequate net positive suction head could cause damage. The valve's postmaintenance test was scheduled after the noted surveillance test, and the surveillance was not intended to check the valve's function. The safety injection train was considered inoperable but available at the time. Licensee personnel entered the noncited violation into the corrective action program as Condition Reports CR-WF3-2008-2280 and CR-WF3-2008-3045.

  This finding was more than minor because it affected both the configuration control and the equipment performance attributes of the Mitigating Systems Cornerstone objective to ensure reliability of the low pressure safety injection system. In addition, this condition, if left uncorrected, would also become a more significant safety concern. Equipment could be damaged without adequate postmaintenance checks prior to operation. Using the NRC Manual Chapter 0609, "Significance Determination Process," Phase 1 Screening Worksheet, the finding was of very low risk significance because it did not: (1) represent a loss of safety function; (2) represent an actual loss of a single train of equipment for more than its Technical Specification allowed outage time; or (3) screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

  This finding had a crosscutting aspect in the area of human performance, associated with the decision-making component, in that, the plant personnel used nonconservative assumptions and chose to use the pump suction valve for system operation prior to verifying that the valve was properly assembled [H.1(b)] (Section 1R19).

- Green. The inspectors identified a noncited violation of 10 CFR 50, Appendix B, Criterion III (Design Control) for an inadequate "pressure locking" design calculation for shutdown cooling Valves SI-405A and SI-405B. Plant engineers also used the calculation to support valve operability following a valve malfunction, which appeared to be caused by pressure locking. Entergy engineers had derived valve bonnet leakage rates (for pressure locking conditions) from local leak rate testing results. However, a national laboratory had already proven the Entergy theory invalid and plant engineers had taken no steps to validate the theory themselves. Finally, in response to an NRC generic letter concerning pressure locking and thermal binding of valves, the licensee engineers' conclusions were based on incorrect facts and improper assumptions. Licensee personnel entered the noncited violation into the corrective action program as Condition Report CR-WF3-2008-4292.

  The failures to perform: (1) an adequate engineering calculation and (2) a valid operability determination were performance deficiencies. This finding was more than minor because it was similar to nonminor finding Example 3.j in NRC Inspection Manual Chapter 0612 Appendix E, "Examples of Minor Issues," in that, there was a reasonable doubt concerning the operability of Valves SI-405A/B. The inspectors utilized NRC Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," to characterize the significance of the issue. Using the worst case scenario of having both SI-405A/B valves inoperable, the finding was of very low safety significance because multiple systems or components would still be available to remove decay heat and respond to a loss of inventory event. These systems included the emergency feedwater system, main feedwater system, auxiliary feed water system, atmospheric dump valves, charging pumps, safety injection tanks, and the high pressure safety injection system. This performance deficiency would not result in any loss of instrumentation needed for safe shutdown and cool down of the plant. The finding had a crosscutting aspect in the area of problem identification and resolution [P.1(c)] because engineers failed to thoroughly evaluate the potential for valve pressure locking. The calculation was completed in 2008 and was indicative of current performance.

B. Licensee-Identified Violations

A violation of very low safety significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and the corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status: The plant began the inspection period on July 1, 2008, at 100 percent power and remained at approximately 100 percent power until September 1 when the plant was shutdown and placed in Mode 4 (Hot Shutdown) in preparation for the arrival of Hurricane Gustav. The plant was returned to 100 percent power on September 11 and has remained there for the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

.1 Summer Readiness of Offsite and Alternate ac Power Systems

a. Inspection Scope

The inspectors verified that plant features, and procedures for operation and continued availability of offsite and alternate ac power systems are appropriate. The inspectors reviewed the licensee's procedures affecting these areas and the communications protocols between the transmission system operator and the Waterford 3 Steam Electric Station to verify that the appropriate information is exchanged when issues arise that could impact the offsite power system. These included: (1) coordination between the transmission system operator and the Waterford 3 Steam Electric Station during an off-normal or emergency event affecting the Waterford 3 Steam Electric Station; (2) explanation of the event; (3) an estimate of when the offsite power system will be returned to a normal state; and (4) notification to the Waterford 3 Steam Electric Station when the offsite power system is returned to normal. In addition, the inspectors verified that the licensee's procedures address measures to monitor and maintain availability and reliability of both the offsite ac power system and the onsite alternate ac power system.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Readiness for Seasonal Extreme Weather Conditions – Site Specific

a. Inspection Scope

The inspectors completed a review of the Waterford-3 readiness for seasonal susceptibilities involving tornados, hurricanes, and other high wind conditions. The inspectors: (1) reviewed plant procedures, the Final Safety Analysis Report, and the Technical Specifications to ensure that operator actions defined in adverse weather procedures and administrative plans, and maintained the readiness of essential systems; (2) walked down portions of the Dry Cooling Towers A and B, transformer yard, battery rooms, and switchgear rooms to ensure that adverse weather protection features were sufficient to support operability, including the ability to perform safe shutdown functions; (3) evaluated operator staffing levels to ensure the licensee could maintain the readiness of essential systems required by plant procedures; and (4) reviewed the corrective action program to determine if the licensee identified and corrected problems related to seasonal conditions.

- August 28, 2008, Preparations for Hurricane Gustav

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdown (71111.04)

a. Inspection Scope

The inspectors: (1) walked down portions of the risk-important system listed below while the other train was out of service and reviewed plant procedures and documents to verify that critical portions of the selected system were correctly aligned; (2) reviewed outstanding work requests; and (3) verified that licensee personnel were identifying and correcting deficiencies through their corrective action program.

- September 2, 2008, 125 VDC Train A

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Complete System Walkdown (71111.04S)

a. Inspection Scope

The inspectors: (1) reviewed plant procedures, drawings, the Final Safety Analysis Report, the Technical Specifications, and vendor manuals to determine the correct alignment of the essential chilled water system Train A; (2) reviewed outstanding design issues, operator workarounds, and open work requests to verify that outstanding issues did not adversely affect the functionality of the system; and (3) verified that licensee personnel were identifying and resolving equipment problems in accordance with corrective action program requirements.

- August 13, 2008, Essential Chilled Water System Train A

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

Quarterly Inspection (71111.05Q)

a. Inspection Scope

The inspectors walked down the four plant areas listed below to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were established for degraded or inoperable fire protection features and that the compensatory measures were commensurate with the significance of the deficiency; and (6) reviewed the Final Safety Analysis Report to determine if the licensee identified and corrected fire protection problems.

- July 23, 2008, Fire Zones RAB 23, 36, 37 and 39
- July 30, 2008, Fire Zones RAB 1E, 6 and 7D
- August 3, 2008, Fire Zones RAB 2, 5 and 16
- September 18, 2008, Dry Cooling Towers A and B

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A)

a. Inspection Scope

The inspectors reviewed licensee programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for the

Train B shutdown cooling system heat exchanger. The inspectors verified that: (1) performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; (2) licensee personnel utilized the periodic maintenance method outlined in EPRI [Electric Power Research Institute] NP-7552, "Heat Exchanger Performance Monitoring Guidelines;" (3) licensee properly utilized biofouling controls; (4) licensee personnel adequately assessed the state of cleanliness of the tubes during heat exchanger inspections; and (5) the heat exchanger was correctly categorized in accordance with 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On July 29, 2008, the inspectors observed training of senior reactor operators and reactor operators in the plant simulator to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The training scenario involved a steam generator tube leak, followed by a component cooling water surge tank level switch failure which prompted the crew to commence a rapid downpower of the plant. The scenario continued with a main turbine manual control failure, followed by two dropped control element assemblies which prompted the operators to trip the plant. Finally, the scenario then included a reactor system cold leg break followed by a failure of Containment Spray Pump A to start.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors reviewed the equipment performance issues listed below to: (1) verify the appropriate handling of structure, system, and component performance or condition problems; (2) verify the appropriate handling of degraded structure, system, and component functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of structure, system, and component issues reviewed under the requirements of 10 CFR 50.65, 10 CFR Part 50, Appendix B, and the Technical Specifications.

- August 11, 2008, Safety-Related tornado and flood door malfunctions
- September 11, 2008, Control room ventilation envelope failures

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

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

Risk Assessments and Management of Risk

a. Inspection Scope

The inspectors reviewed the three assessment activities listed below to verify: (1) performance of risk assessments when required by 10 CFR 50.65(a)(4) and licensee procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that the licensee recognized, and/or enters as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; (4) the licensee properly controlled emergent work; and (5) the licensee identified and corrected problems related to maintenance risk assessments.

- August 6, 2008, Dry cooling Tower B sump Pump B replacement
- August 15, 2008, Reactor trip circuit breaker tests
- September 11, 2008, High pressure safety injection Pump B fail to start

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plant status documents, such as operator shift logs, emergent work documentation, deferred modifications, and standing orders, to determine if an operability evaluation was warranted for degraded components;

(2) referred to the Final Safety Analysis Report and design-basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any Technical Specifications; (5) used the significance determination process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

- July 28, 2008, Containment fan Cooler B low component cooling water flow

- July 30, 2008, Containment fan Cooler D boric acid accumulation

- August 21, 2008, RCS hot leg input to core protection calculator Channel D reading four degrees higher than hot Legs A, B, and C

- August 24, 2008, Low pressure safety injection Pump A start with suction isolation valve shut

- September 3, 2008, 125 VDC Train B following repairs to loose battery connection

- September 6, 2008, Valve SI-405B apparently pressure-locked when operators attempted to initiate shutdown cooling operations

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

.1 Inadequate Operability Assessment for Valve Pressure Locking:

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III (Design Control) for an inadequate "pressure locking" design calculation involving shutdown cooling Valves SI-405A and SI-405B. Plant engineers also used the calculation to support valve operability following a valve malfunction, which appeared to be caused by pressure locking. Entergy engineers had derived valve bonnet leakage rates (for pressure locking conditions) from local leak rate testing results. However, a national laboratory had already proven the Entergy theory invalid and plant engineers had taken no steps to validate the theory themselves. Finally, in response to an NRC generic letter concerning pressure locking and thermal binding of valves, the licensee's conclusions were based on incorrect facts and improper assumptions.

Description.

Background: Valves SI-405A/B are opened to place shutdown cooling and low-temperature/over-pressure protection (Valves SI-406A/B) in service. Valves SI-405A/B are in series between Valves SI-401A/B and SI-407A/B, which are also opened for shutdown cooling operations.

In the past refueling outage (RF-15, Spring 2008), plant craftsmen replaced Valves SI 405A/B hydraulic actuators with air-operated actuators. Plant engineers specified the modification to resolve performance problems with the hydraulic units. However, the behavior of the air-operated actuators was fundamentally different than that of the hydraulic model. Specifically, the hydraulic actuators provided up to the maximum amount of stem thrust starting at the beginning of the opening valve stroke, when thrust demands were greatest. In contrast, the air-operated actuators gradually increased valve stem thrust as air on the top of the actuator piston was vented off. If a load existed on the stem (typically the case, i.e. packing and unwedging), the valve would not initially move when commanded to open. It would remain closed until the actuator developed sufficient thrust to move the valve from its seat. Once off the seat, the valve disc would pop to a midposition and then continue to the open position at the normal speed.

On May 14, 2008, during post-installation testing, a plant engineer observed that Valve SI-405A did not initially move on the first two attempts. As noted in CR WF3-2008-02326, approximately 2 minutes following the second attempt, the valve opened with a: ". . . loud abrupt sound and shaking of the surrounding area. The valve coupling on the stem immediately moved approximately one half of the travel distance following the loud noise."

Just prior to the valve operation, operators had pressurized the shutdown cooling piping system to approximately 1700 psig. The valve had pressure locked following the test. Pressure locking occurs when elevated residual pressure is trapped in a valve's bonnet and prevents the valve from opening. The NRC has issued several industry wide generic communications concerning pressure locking, most notably Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," dated August 22, 1995. Entergy was required to respond to the generic letter, documenting their plans to address these generic concerns. Entergy engineers had concluded that pressure locking of these valves would not occur. Specific deficiencies with the Entergy generic letter response are provided later in this report.

Intersystem Loss-of-Coolant Event: On September 1, plant operators shut down Waterford-3 in preparation for Hurricane Gustav. During initiation of Train B shutdown cooling operations, control room operators identified and responded to an intersystem loss-of-coolant event that lasted approximately 4 minutes. About 800 gallons of reactor coolant was lost through low-temperature over-pressure protection Relief Valve SI-406B. The relief valve unexpectedly opened following an apparent malfunction of Valve SI-405B.

While placing Train B shutdown cooling in service, Valve SI-405B had failed to initially move when the control room operator repositioned the valve's control room switch to open. The indication showed full closed for about 12 minutes. The valve indication then showed midposition, which was followed by full open indication. The air-operated valve normally had a 5 to 6 minute stroke time. Licensee personnel determined that the valve was stuck in its seat until the valve actuator developed sufficient thrust to move the valve disc. Then, the valve popped to almost the full open position.

Entergy engineers determined that the sudden motion of Valve SI-405B created a pressure transient in the residual heat removal piping. Since system pressure was about 350 psig at the time, system pressure combined with the pressure transient that resulted from the opening of Valve SI-406B exceeded the setpoint of Relief Valve SI-406B, which was approximately 430 psia. Once lifted, relief valves will not normally seat at their set pressure but will close at a pressure somewhat below the setpoint. Since system pressure was already relatively high, the valve did not immediately reseat. Operator action was necessary to stop the leakage by closing Valve SI-401B. This was the preferred isolation valve because of the slower closing time of Valve SI-405B.

At the end of the inspection period, the inspectors were still evaluating information related to the intersystem loss-of-coolant event. The inspectors provided questions to licensee personnel concerning the valve actuator design change package. This is an URI pending further evaluation by the NRC inspectors: URI 05000382/2008004-01, intersystem loss-of-coolant event.

Operability Assessment (Documented in CR-WF3-2008-4161): In response to the event, licensee personnel performed an operability assessment to address two potential concerns. First, they addressed the setpoint of Relief Valve SI-406B to ensure that the setpoint had not drifted too low, potentially creating the potential for similar future events. Licensee personnel determined that the setpoint was appropriate and that the valve remained operable. However, the behavior of Valve SI-405B presented the potential for continued intersystem loss-of-coolant events. Subsequently, engineers completed procedure changes to re-sequence the opening of the shutdown cooling valves. Valves SI-405A/B would be initially opened only when the flow paths from the reactor coolant system to Relief Valves SI-406A/B were isolated by Valves SI-401A/B. Once opened for the first time, the pressure locking condition should be cleared and subsequent valve repositioning can be accomplished without the risk of a challenging pressure transient. The inspectors found that the evaluation and corrective measures to address this point were acceptable.

The second operability point involved the failure of Valve SI-405B to open on command. If either Valve SI 405A/B failed to open, the associated train of shutdown cooling would be inoperable. Licensee personnel concluded that the valves were operable because pressure locking was not the failure mechanism observed. Engineers relied on results from Calculation ECM 08-002, "Susceptibility of SI-405A/B to Pressure Locking," Revision 0, dated February 18, 2008, which determined that valve bonnet pressure would decay within 1.8 hours following plant shutdown and the initiation of plant depressurization. Since initiation of depressurization had occurred much more than 1.8 hours prior to the event, Entergy engineers concluded that pressure locking was not a factor. In addition, the engineers determined that the amount of valve stem thrust required to move the disk was consistent with that calculated to unwedge the valve disc from its seat. Calculation ECM91-076, Revision 3, provided a maximum unseating thrust of 26,855 pounds. Therefore, Entergy engineers determined that the delay in unseating the valve was due to the normal valve disc unwedging load.

Inadequate Basis for Operability: The inspectors determined that the licensee's basis for operability was inadequate.

- First, engineers had no basis for the conclusion that the thrust required to open Valve SI-405B during the September 1 event was simply the normal unwedging load. The inspectors noted that the valve actuator could provide up to 49,457 pounds of thrust to open the valve. Since the valve did not move for approximately 12 minutes, it was unclear how much thrust was applied to the valve stem during this period. The valve only had a 5 to 6 minute stroke time and it was possible that the entire 49,457 pounds of thrust was being applied. In addition, the maximum calculated unwedging thrust is normally a "bounding" design value (26,855 pounds in this case). The actual valve unwedging thrust obtained during diagnostic testing was much lower, at approximately 7700 pounds. Further, these unwedging thrusts have not been shown to significantly change over time. To illustrate this point, the inspectors asked licensee personnel to demonstrate (from the large body of diagnostic system test results at Waterford-3) that unwedging thrusts changed significantly over time for any flexible wedge gate valve. Licensee personnel reviewed the test results and provided no examples to support their contention. Therefore, the inspectors determined that an additional thrust component (besides the normal expected unwedging thrust) was at work. Possible additional thrust loads included: (1) unanticipated differential pressure; (2) pressure-locking; and/or (3) thermal binding.

- Second, Calculation ECM-08-002, which was the primary justification for concluding that pressure locking had not occurred, was inadequate. Specifically, the calculation used an inappropriate method to determine the valve bonnet leak rate. Plant engineers derived the bonnet leak rate from local leak rate test results (a type of differential pressure test). The inspectors noted that the Waterford-3 engineers had performed no testing or engineering analysis to validate the theory behind the calculation. Conversely, the inspectors noted that the Idaho National Engineering Laboratory (Report INEL-96/00161, dated April 1996, page 7) had performed valve pressure-locking testing to specifically evaluate the merits of this theory and found that there was no discernable relationship between differential pressure leak rates and bonnet leak rates under pressure locking conditions.

The inspectors determined that the inadequate pressure-locking calculation was a violation of 10 CFR Part 50, Appendix B, Criterion III (Design Control). This requirement allows licensee personnel to use calculations to verify or check the adequacy of design. The failure to perform an adequate calculation, however, constitutes failure to meet the requirement. In response to the inspectors concerns, Entergy personnel performed a second operability determination. The second operability determination is discussed later in this report.

Generic Letter 95-07 Supplemental Response: On June 24, 1996, the NRC submitted to Entergy, Waterford-3, a request for additional information concerning Entergy's Generic Letter 95-07 response. The request concerned three pairs of valves that the NRC believed had the potential to pressure lock or thermally bind. One of those pairs was the SI-405A/B valves. In addition, on May 13, 1999, NRC staff provided additional questions to the licensee during a followup phone conversation. In short, licensee personnel were asked to justify why the subject valves would not be subjected to pressure-locking or thermal binding.

Licensee personnel responded to the request for information in a letter to the NRC dated June 17, 1999. For the SI-405 A/B valves, Entergy engineers took the following positions:

- Concerning the potential for higher containment temperatures following a small break loss-of-coolant accident to result in increased valve bonnet temperatures and pressures, Entergy engineers specified: "Following a small break loss-of-coolant accident, it is expected that the temperature at . . . SI-405A/B will either remain constant or decrease prior to the initiation of shutdown cooling."

  To support this conclusion, plant engineers had reasoned that, as reactor coolant system temperature decreased (as part of the normal shutdown), valve bonnet temperatures would also decrease.

  In addition, plant engineers had made temperature measurements on the exterior of the Valve SI-405A/B bonnets during a plant shutdown. Technicians found temperatures to be approximately 218°F, but the temperature dropped more than 3°F over a 5 hour 41 minute period. Since containment temperatures for small break loss-of-coolant accidents are not expected to exceed this temperature for a significant period of time, Entergy engineers concluded that the containment atmosphere would not provide a heating source for the valve bonnets. They also believed that the small decrease in temperature on the valve external was indicative of a general cooling inside the valve bonnets.

  The inspectors found that the licensee engineers had no technical basis to support their conclusion. Licensee personnel had performed no testing nor analysis to show that temperatures within the bonnets of Valves SI-405A/B would remain constant or decrease prior to the initiation of shutdown cooling during a small break loss-of-coolant accident. Since there was only a small amount of flow through Valves SI-405A/B when all the shutdown cooling isolation valves were closed, it was not reasonable to assume that the reduction in reactor coolant system temperature would necessarily translate into meaningful temperature reduction within the bonnets of Valves SI-405A/B.

  The inspectors also considered flawed the Entergy engineers' conclusion that increasing containment atmosphere temperatures would not result in the heatup of Valve SI-405A/B bonnets. The engineers failed to consider that the overall heat transfer equilibrium would change during a small break loss-of-coolant accident and that heat flow from the valves to the containment atmosphere would be reduced. Since normal containment temperature was about 110°F, and post small break loss-of-coolant accident containment temperature was expected to reach 215°F, the differential temperature between the external valve bonnets and the containment would reduce substantially. This reduction in the differential temperature would result in less heat transfer from the valves to containment and would result in overall higher temperatures within the valve bonnets.

  - The Entergy engineering response stipulated: Operating history has shown that these valves have never failed to open during normal shutdown cooling operations.

While the subject statement was made in the thermal binding section of the Entergy response, the conclusion was important to the pressure-locking considerations as well. A good valve operational history spoke for itself.

The inspectors identified that this statement was not correct. Specifically, the subject valves had failed to open on two occasions prior to the submittal date - once when attempting to open Valve SI-405B for shutdown cooling operations on September 19, 1998 (CR-WF3-1998-01241 and 01243) and once when attempting to open SI-405A prior to initiation of low temperature over-pressure protection (a similar operating condition) on June 11, 1995 (CR WF3 1995 00477). Licensee engineers had attributed these particular failures to gas coming out of solution in the hydraulic pump, which decreased the pump's discharge pressures. However, it was unclear from the documentation if pressure locking or thermal binding had contributed to the failures. Overall, Valves SI-405A/B have experienced a relatively large number of operational problems. Other problems included:

- April 12, 1997, both SI-405A and B exceeded their maximum open stroke times (CR-WF3-1997-00852).

- October 27, 2000, SI-405B would not stroke open. The motor overload had tripped. Once reset, the valve opened but very slowly (stroke time was almost 18- minutes). The motor was not developing sufficient discharge pressure (CR-WF3-2000-01347).

- March 23, 2002, the plant entered an Alert because operators could not open both Valves SI-405A and B for shutdown cooling operations. Entergy engineers attributed the failures to thermal binding. Operators had flooded the piping near the valves with cold water (to preclude void formation) just prior to the event (CR-WF3-2002-00468).

- April 5, 2002, Valve SI-405B failed to open when aligning the system for shutdown cooling operations (MAI 434819).

- October 20, 2003, Valve SI-405B exceeded its maximum allowed stroke time during inservice testing (CR-WF3-2003-02991).

- November 26, 2006, Valve SI-405B did not stroke open within 15 minutes. A time delay interlock commanded the valve to close because it was not full open within the 15 minute limit (CR-WF3-2006-03610).

- October 9, 2007, the Valve SI-405B stroke time exceeded its maximum allowed value. The followup stroke was satisfactory (CR-WF3-2007-03553).

- April 27, 2008, Valve SI-405B stroked open in excess of its maximum permitted stroke time (CR-WF3-2008-01671). The second stroke was satisfactory.

- May 14, 2008, Valve SI 405A did not open because of pressure locking. Operators had conducted a system pressure test prior to the valve repositioning (CR-WF3-2008-02326).

The inspectors noted that several of the valve malfunctions could have been caused, at least in part, by pressure locking or thermal binding. Each hydraulic actuator pump had a pressure relief valve at the discharge. If Valve SI-405A/B was stuck closed, the hydraulic pump relief valve could lift and the pump discharge pressure would be substantially less while the relief valve remained open. The relief valve may not reseat until the hydraulic pump was secured. This particular scenario would be consistent with slow valve operation and/or valve failure. Further, once finally repositioned, the hydraulic pump would secure and the relief valve would reseat. Followup strokes would show improvement (since the valve operation would serve to clear the pressure-locking or thermal binding condition). The licensee did not collect sufficient data from the valve malfunctions to rule out pressure locking and thermal binding as possible contributors. As demonstrated by the May 14, 2008, event, the valves were susceptible to pressure-locking. The valves also appeared sensitive to thermal binding, based of the March 22, 2002, event.

In conclusion, the inspectors determined that licensee personnel had inadequate basis to conclude that Valves SI-405A/B would not be subjected to pressure-locking and thermal binding operational conditions.

Analysis. The failures to perform: 1) an adequate engineering calculation; and 2) a valid operability determination were performance deficiencies. This finding was more than minor because it was similar to nonminor finding Example 3.j in NRC Inspection Manual Chapter 0612 Appendix E, "Examples of Minor Issues," in that there was a reasonable doubt concerning the operability of Valves SI-405A/B. The inspectors utilized NRC Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," to characterize the significance of the issue. Using the worst case scenario of having both Valves SI-405A/B inoperable, the finding was of very low safety significance because multiple systems or components would still be available to remove decay heat and respond to a loss-of-inventory event. These systems included the emergency feedwater system, main feedwater system, auxiliary feed water system, atmospheric dump valves, charging pumps, safety injection tanks, and the high-pressure safety injection system. This performance deficiency would not result in any loss of instrumentation needed for safe shutdown and cool down of the plant. The finding had a crosscutting aspect in the area of problem identification and resolution [P.1(c)] because engineers failed to thoroughly evaluate the potential for valve pressure locking. The calculation was completed in 2008 and was indicative of current performance.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III (Design Control), requires, in part, that measures be established to provide for the verifying (or checking) the adequacy of design. These measures may include calculations. The licensee used Calculation ECM 08-002 to demonstrate the adequacy of the safety injection Valve SI-405A/B design. Specifically, the calculation evaluated the potential for pressure locking of the subject valves. Contrary to the above, as of September 1, 2008, the design control measures for Valves SI-405A/B were inadequate, in that Calculation ECM 08-002 used an invalid method to determine the susceptibility of the valves to pressure-locking conditions. Because the violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report CR-WF3-2008-04292, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000382/2008004-02, inadequate pressure locking calculation.

.2 Second Operability Determination for Valves SI-405 A/B

Introduction. The inspectors opened an unresolved item to address potential concerns associated with the licensee's second operability determination for safety injection Valves SI-405A/B.

Discussion. In response to the inspectors' continued operability concerns, Entergy personnel performed a second operability determination, as documented in CR-WF3-2008-04294. The licensee contracted with MPR Associates, an engineering firm, to assess the potential for pressure locking of Valves SI-405A/B. MPR produced a calculation that concluded that the air-operated actuators could overcome the thrust requirements for a bounding pressure locking event. The MPR calculation assumed that bonnet pressure was 2250 psig (normal plant operating pressure) and that no pressure existed in the upstream and downstream piping sections. However, the inspectors observed the following apparent inconsistencies with the calculation:

- Waterford-3 engineers used, and relied upon, the calculation in their operability assessment but neither Waterford-3 engineers nor MPR had performed a design verification of the calculation. The inspectors noted that Procedure EN-DC-126, "Engineering Calculation Process," Revision 1, stated, in part: Calculations prepared in accordance with this procedure to support a reasonable expectation of operability do not need to be design verified.

This procedure stipulation appeared inconsistent with the Waterford-3 "Quality Assurance Program Manual," Revision 18, dated April 15, 2008, Section B.3.d (Design Verification) which stated, in part: Independent design verification is to be completed before design outputs are used by other organizations for design work. . . In all cases [emphasis added], the design verification is to be completed before relying on the item to perform its function.

- The conditions assumed in the calculation were substantially more demanding than could have been experienced during known pressure locking event on May 14, 2008, (Valve SI-405A) but the calculation concluded that the actuator had sufficient thrust to overcome even more demanding pressure locking conditions. Specifically, the calculation assumed a bounding 2250 psig in the bonnet, which exceeded the worst case valve bonnet pressure of 1700 psig during the May 14, 2008, pressure locking event. Therefore, it appeared that MPR had concluded that the May 14, 2008, event could not have occurred. Entergy and MPR engineers stated that they had not evaluated the May 14, 2008, event. Entergy and MPR engineers were working to resolve the apparent inconsistency at the close of the inspection period.

- Validation analysis performed to support the MPR pressure locking calculation method, "Pressure Locking and Thermal binding Evaluation of EPRI MOVs [Motor-Operated Valves]," showed that the method appeared conservative if a high valve friction coefficient (0.61) was used, but this high friction coefficient was not used in the Waterford-3 SI-405A/B analysis. The validation calculation also provided case studies involving lower valve friction coefficients (called the "best-fit" coefficients of friction), but the method under predicted required actuator thrust for 38 percent of the examples when these values were used. In addition, when valve bonnet pressures were greater than 500 psig, the method under predicted required valve thrust in 75 percent of the cases and it under predicted required valve thrust in all three instances where bonnet pressure was greater than 1000 psig.

NOTE: EPRI did not assess pressure locking as part of their MOV validation testing program. However, six of the valves demonstrated pressure locking symptoms (retaining bonnet pressure). In addition, two of these valves had a repeated instance of bonnet pressure retention. MPR validated their calculation method against this data.

- Also concerning the validation analysis, the "best-fit" friction coefficients were generally in the midrange of friction coefficients observed for any given valve tested as part of the EPRI program. For example, for Valve MOV 15, MPR had assumed a valve friction coefficient of 0.2. The actual EPRI test results stated that the value was approximately 0.18 to 0.2 but the test data ranged from 0.17 to 0.3. A second example involved Valve MOV 24. The friction coefficient used in the validation analysis was 0.53. The EPRI report stated that coefficient of friction stabilized at about 0.44 but data ranged between 0.32 and about 0.6. Similar observations were made for all six test valves and all eight test cases. It was not clear how MPR had selected the "best-fit" friction coefficients from the EPRI test data.

- The MPR calculation, "Pressure Locking Evaluation of SI-405A(B)," dated September 11, 2008, provided unexpected results. MPR performed case studies for the Waterford-3 valves using different valve friction coefficients (0.2, 0.35, and 0.5). All other variables were held constant. The inspectors noted that the required thrust for the 0.5 valve friction coefficient (41,787 pounds of thrust) was less than that calculated for the 0.35 friction coefficient (42,218 pounds of thrust). Normally, as with a friction coefficient (with all other variables held constant) the required thrust would also be higher. MPR had not answered all questions related to this inconsistency at the close of the inspection.

For all of the above potential concerns, the licensee was working to obtain additional information to support their conclusions. This is considered an URI pending NRC review of the licensee provided information: URI 05000382/2008004-03, operability of safety injection Valves SI-405A/B.

Analysis. No significance determination was performed for this URI. If the NRC determines that a valid finding or violation occurred, a significance determination will be performed at that time.

Enforcement. No enforcement is recommended at this time. Enforcement will be considered when closing the URI.

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17)

a. Inspection Scope

This inspection procedure is a combination of two previous baseline inspection procedures: (1) Evaluations of Changes, Tests, or Experiments (71111.02); and

(2) Permanent Plant Modifications (71111.17B). The procedure is now performed on a triennial basis and requires a minimum sampling of 5 permanent plant modifications, 6 evaluations required by 10 CFR 50.59, and 12 changes, tests, or experiments that were screened by the licensee's program as not requiring an evaluation.

The objectives of this procedure are to verify that evaluations were performed in accordance with the requirements of 10 CFR 50.59; that the design bases, licensing bases, and performance capability of structures, systems, and components have not been degraded through modifications; and that design and license basis documentation affected by and used to support changes, have been adequately updated and reflect the design and license basis of the facility after the change has been made.

The inspection was performed with an in-office review and preparation period, followed by an onsite review period at Waterford The inspectors reviewed licensee procedures for engineering change development, installation, testing, and closure. Procedures for process applicability determination and 10 CFR 50.59 program review were also reviewed.

The inspectors reviewed nine evaluations and supporting documentation, including drawings, calculations, Final Safety Analysis Report, and Technical Specifications to confirm the licensee's conclusions that the changes would not require application for a license amendment. The evaluation samples were chosen based on risk significance, safety significance, and complexity. The listing of evaluations reviewed is included in the list of documents reviewed.

The inspectors reviewed 15 examples of screenings for which the licensee had concluded that evaluations were not required. The review confirmed that the licensee's conclusions were correct and consistent with the requirements of 10 CFR 50.59. The screenings reviewed are listed in the list of documents reviewed.

The inspectors evaluated 10 permanent plant modification packages. The modifications were reviewed for adverse effects on system availability, reliability, and functional capability. Documents reviewed included calculations, modification design and change packages, drawings, corrective action documents, and applicable sections of the Final Safety Analysis Report, Technica;Specification, and design basis documents. The inspectors reviewed post maintenance test documentation to ensure adequacy in scope and conclusion. The modifications reviewed are listed in the list of documents reviewed.

The inspectors reviewed a sample of recent licensee condition reports related to the 10 CFR 50.59 and the permanent plant modification processes to determine whether the licensee had identified problems and entered them into the corrective action program at the appropriate threshold. Condition report documents reviewed are listed in the list of documents reviewed.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

Permanent Modifications

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report, plant drawings, procedure requirements, and the Technical Specifications to ensure that the permanent modification listed below was properly implemented. The inspectors: (1) verified that the modification did not have an affect on system operability/availability; (2) verified that the installation was consistent with modification documents; (3) ensured that the post installation test results were satisfactory and that the impact of the temporary modification on permanently installed structures, systems, and components were supported by the test; (4) verified that the modification was identified on control room drawings and that appropriate identification tags were placed on the affected drawings; and (5) verified that appropriate safety evaluations were completed. The inspectors verified that licensee identified and implemented any needed corrective actions associated with temporary modification.

- September 6, 2008, Modification to motor operators to safety injection isolation Valves SI-401A/B to allow the valves to open in two stages

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

b. Inspection Scope

The inspectors reviewed the six postmaintenance test activities of risk significant systems or components listed below. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design-basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly realigned, and deficiencies during testing were documented.

- July 21, 2008, Retest of diesel fire Pump No. 2 following lantern ring replacement and pump repack

- July 22, 2008, Test of safety injection Valve SI-109A following valve replacement

- July 31, 2008, Retest of control room emergency filtration Unit A following adjustments to limit switch trip settings

- September 3, 2008, 125 VDC Train B battery tests following maintenance to retighten battery post connection

- September 7, 2008, Safety injection isolation Valves SI-401A/B following modifications to the motor operator

- September 10, 2008, Replacement of dry cooling Tower No. 2 sump pump discharge check Valve SP-3271

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

Introduction. The inspectors identified a noncited violation of Technical Specification 6.8.1.c (Procedures) for the failure to open the Train A low pressure safety injection pump suction valve prior to pump operation during a surveillance. The butterfly valve was installed 90 degrees out of position and was closed when operators believed it was open. After starting the pump, operators observed loud noises coming from the unit and secured it 8 minutes later. Pump operation without adequate net positive suction head could cause damage. The safety injection train was considered inoperable but available at the time.

Description. During Refuel 15 outage activities, low pressure safety injection (LPSI) Pump A suction isolation Valve SI-109A was replaced due to valve leakage. Valve SI-109A is a quarter turn, 20-inch butterfly valve. The actuator was removed from the old valve assembly and was reused on the new valve assembly, since no problems had been identified with the valve actuator. Following Valve SI-109A replacement and actuator installation, the valve was stroked to consolidate packing, but valve stroke position verification was not documented.

On May 13, 2008, licensee personnel performed Procedure OP-903-115, “Train A Integrated Emergency Diesel Generator - Engineering Safety Features Test,” Revision 9, to test the Emergency Diesel Generator A and its associated engineered safety features train. Prerequisite Step 2.10 requires that safety injection be aligned to allow operation of LPSI Pump A on recirculation. In an attempt to ensure that this prerequisite step was met, a LPSI Train A valve lineup was performed. This valve lineup requires Valve SI-109A to be in the open position. Two operators used the standard, approved method to verify Valve SI-109A to be in the open position, by checking for valve motion in the closed direction, then returning the valve to its full open seat. At the time this action was performed, Valve SI-109A was still in a maintenance state, with postmaintenance testing yet to be completed.

On May 13, when LPSI Pump A was started per Procedure OP-903-115, the pump “made loud noises” and was secured after 8 minutes. Trouble shooting determined that Valve SI-109A was in the closed position, even though the operator verified the valve position indicated open. With Valve SI-109A in the closed position, LPSI Pump A is not aligned to allow the recirculation operation required by Procedure OP-903-115.

The subsequent apparent cause evaluation determined that the old valve and actuator were removed while in the open position. The new valve was installed in the closed position, but the old actuator was installed without changing it from the old valve's position (open) to match the new valve's position (closed). As a result, the actuator indicated open, when actual valve position was closed. This was further complicated by the fact that the new valve disc rotated counter clockwise to open, while the original valve disc rotated clockwise to open. The combination of these circumstances allowed the operators performing the valve lineup to obtain erroneous results when performing the check.

Licensee management discussed waiting for the postmaintenance testing to be performed on Valve SI-109A, prior to performing Procedure OP-903-115, but opted to declare the system "available" and conduct the test. A valve lineup was relied upon to ensure that Valve SI-109A was in the correct position, but without any postmaintenance testing to ensure that the valve would behave as expected, the valve lineup alone was insufficient to ensure its position and ultimately, procedural compliance.

Analysis. The failure to perform the necessary task to ensure compliance with plant procedures was a performance deficiency. This finding was more than minor because it affected both the configuration control and the equipment performance attributes of the Mitigating Systems Cornerstone objective to ensure reliability of the low pressure safety injection system. Using the NRC Manual Chapter 0609, "Significance Determination Process," Phase 1 Screening Worksheet, the finding was of very low risk significance because it did not: (1) represent a loss of safety function; (2) represent an actual loss of a single train of equipment for more than its Technical Specification allowed outage time; nor (3) screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding had a crosscutting aspect in the area of human performance, associated with the decision making component, in that, the plant personnel used nonconservative assumptions and chose to use the pump suction valve for system operation prior to verifying that the valve was properly assembled [H.1(b)].

Enforcement. Technical Specification 6.8.1.c, requires that the licensee correctly implement surveillance procedures on safety-related equipment. The failure to ensure that Procedure OP-903-115, Prerequisite Step 2.10 was met prior to implementation of the procedure is a violation of this requirement. Because this violation was of very low safety significance and was entered in the corrective action program as Condition Reports CR-WF3-2008-2280 and CR-WF3-2008-3045, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000382/2008004-04, failure to follow integrated EDG test procedure.

1R20 Unplanned Outage (71111.20)

a. Inspection Scope

The inspectors evaluated plant personnel's activities related to the unplanned outage conducted from September 1 to 11, 2008, following the planned plant shutdown in preparation for Hurricane Gustav. This was done to verify that plant personnel had: (1) developed mitigation strategies for losses of key safety functions, and adhered to operating license and Technical Specification requirements that ensured defense-in-depth; and (2) ensured areas not accessible during at power operations were inspected to verify that safety-related and risk-significant safety system components were maintained in an operable condition.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Routine Surveillance Testing

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report, procedure requirements, and the Technical Specifications to ensure that the six surveillance activities listed below demonstrated that the structures, systems, and components tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method demonstrated Technical Specification operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of ASME Code requirements; (12) updating of performance indicator data; (13) engineering evaluations, root causes, and bases for returning tested structures, systems, and components not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciators and alarms setpoints. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- July 14, 2008. Train B emergency diesel generator and subgroup relay operability test
- July 31, 2008, Train A main steam system quarterly inservice test valve tests
- August 14, 2008, Train A dry cooling tower diesel-driven sump pump operability test
- August 27, 2008, Train A high pressure safety injection pump operability test
- August 27, 2008, Channel D excore nuclear instrument functional test
- September 11, 2008, Safety injection containment isolation Valves SI-405 A/B inservice test

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

Emergency Preparedness Drill Observation

a. Inspection Scope

For the below listed simulator-based evolution contributing to drill/exercise performance and emergency response organization performance indicators, the inspectors: (1) observed the training evolution to identify any weaknesses and deficiencies in classification, notification, and protective action requirements development activities; (2) compared the identified weaknesses and deficiencies against licensee-identified findings to determine whether the licensee is properly identifying failures; and (3) determined whether licensee performance is in accordance with the guidance of the Nuclear Energy Institute (NEI) 99-02, “Voluntary Submission of Performance Indicator Data,” Revision 5, acceptance criteria.

- July 16, 2008: The inspectors observed a simulator-based training which began with a reactor coolant system leak and containment area radiation monitor alarms, requiring a controlled plant shutdown and an Alert declaration. This was followed by a loss of component cooling water Pump A and charging Pump A. A subsequent reactor coolant system rupture lead to safety injection actuation, main steam isolation, containment isolation, emergency feedwater actuation, and containment spray actuation, in which containment spray Pump A fails to start. Later, a containment penetration failure causes a General Emergency to be declared.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstone: Initiating Events

a. Inspection Scope

Inspectors sampled licensee submittals for the performance indicator listed below for the period October 2007 through September 2008. The definitions and guidance of NEI 99-02, “Regulatory Assessment Indicator Guideline,” Revision 5, were used to verify the licensee’s basis for reporting each data element in order to verify the accuracy of performance indicator data reported during the assessment period. The inspectors reviewed licensee event reports, monthly operating reports, and operating logs as part of the assessment. Licensee performance indicator data were also reviewed against the requirements of Procedure EN-LI-114, "Performance Indicator Process," Revision 2.

- July 31, 2008, Emergency AC power system

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a daily screening of items entered into the license's corrective action program. This assessment was accomplished by reviewing condition reports and event trend reports and attending daily operational meetings. The inspectors: (1) verified that equipment, human performance, and program issues were being identified by the licensee at an appropriate threshold and that the issues were entered into the corrective action program; (2) verified that corrective actions were commensurate with the significance of the issue; and (3) identified conditions that might warrant additional followup through other baseline inspection procedures.

b. Findings

No findings of significance were identified.

.2 Selected Issue Follow-up Inspection

a. Inspection Scope

In addition to the routine review, the inspectors selected the one issue, listed below, for a more in-depth review. The inspectors considered the following during the review of the licensee's action: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

- August 4, 2008, Emergency feedwater Train A unisolable line break due to external corrosion and extent of condition

b. Findings

No findings of significance were identified

4OA3 Event Followup (71153)

.1 Hurricane Gustav Event

a. Inspection Scope

Following Hurricane Gustav, the NRC inspectors and representatives of the Federal Emergency Management Agency assessed the readiness of Waterford 3 Steam Electric Station to restart following the plant shutdown that occurred on June 1, 2008, in preparation for the hurricane. This assessment was implemented in accordance with NRC Inspection Manual Chapter 1601, "Communication and Coordination Protocol for Determining the Status of Offsite Emergency Preparedness Following a Natural Disaster, Malevolent Act, or Extended Plant Shutdown." Specific activities included an assessment of both the onsite and offsite emergency preparedness capabilities and evaluating the readiness of the licensee's systems, structures, components, and staffing levels to support safe plant operations.

To evaluate the condition of the onsite emergency preparedness infrastructure the inspectors determined the status of the following:

- Communication circuits between the licensee and the offsite authorities
- Licensee's emergency response facilities
- Licensee's ability to staff critical emergency response positions
- Environmental monitoring
- Meteorological monitoring
- Evacuation routes to and from the plant site
- Emergency sirens
- Structures, systems, and components needed for emergency action level classifications

Additionally, the inspectors reviewed the licensee's assessment of emergency preparedness program deficiencies. For those deficiencies identified the inspectors reviewed documentation, interviewed personnel, and verified by observations that the compensatory measures in place were adequate.

b. Findings and Observations

After review, the inspectors concluded that both the onsite and offsite emergency preparedness infrastructure were adequate to support plant restart and provide reasonable assurance that protective measures could be taken to protect public health and safety in the event of a radiological emergency. Additionally, the inspectors concluded that all systems, structures, components, and staffing levels were adequate to support safe operation of the plant following restart.

.2 (Closed) Licensee Event Report (LER) 05000382/2008-003-00: Movement of Fuel Assembly in Reactor Vessel with Fuel Handling Machine Inoperable

On May 18, 2008, at about 4:59 p.m. with the plant shut down in Mode 6 for Refuel 15, Fuel Assembly LAY405 was raised within the reactor vessel without the refueling machine automatic overload cut-off protection enabled, a condition prohibited by Technical Specification 3.9.6. The LER was reviewed by the inspectors and the licensee identified noncited violation is documented in this report, in Section 4OA7, licensee identified violations. This LER is closed.

4OA5 Other

Quarterly Resident Inspectors Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors performed observations of security force personnel and activities to ensure that the activities were consistent with Waterford 3 Steam Electric Station security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

Documents reviewed by the inspectors are listed in the attachments.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meeting Summary

On July 15, 2008, the regional inspectors presented the inspection findings related to the evaluations of changes and plant modifications to Mr. K. Walsh and other members of his staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

On September 30, 2008, the resident inspectors presented the quarterly inspection results to Mr. K. Walsh, Site Vice-President, and other members of licensee management at the conclusion of the inspection. 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 two examples of a violation of very low safety significance was identified by the licensee and was a violation of Technical Specification 6.8.1.b, which states that written procedures shall be established, implemented, and maintained for refueling operations. This violation of NRC requirements meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as a noncited violation.

- Section 3.33 of licensee Refueling Procedure RF-005-002, “Refueling Equipment Operation,” states that using the key override to move the refueling machine hoist in the outward direction with a fuel assembly in the core region would require entering the limiting condition for operation for Technical Specification 3.9.6. Technical Specification 3.9.6, requires suspending movement of fuel assemblies when the refueling mast overload cut off limit of less than or equal to 3350 pounds was unavailable. Contrary to the above, on May 18, 2008, the overload cut off limit was unavailable and operators placed the refueling machine in key override and moved a fuel bundle in the outward direction. The operators did not enter Technical Specification 3.9.6. The operators were in the process of moving fuel when the refueling machine computer had "locked-up." In an effort to reboot the computer, licensee personnel placed the refueling machine in key override, which bypassed the refueling equipment interlocks. The intent was to place the mast in a position that would allow the computer to be rebooted. The personnel failed to realize that the actions were not permitted by Technical Specification and plant procedures.

- Section 3.2 of licensee Refueling Procedure RF-005-002 states, that while fuel movement is in progress, PEER check/verifications are required for grapple operation of the refueling machine including verification of proper Z-coordinate, and weight verification when raising the hoist. Section 3.31 states that operation of fuel handling equipment with an interlock bypassed raises the risk of damaging fuel assemblies and equipment. Finally, a caution statement in Section 5.1.8 states that if the refueling machine load varies more than 100 pounds during withdrawal of a fuel assembly, fuel movement should be terminated. Contrary to this, on May 18, 2008, operators proceeded to lift a fuel assembly without verifying that the load did not vary by more than 100 pounds. The load varied by about 1400 pounds. The senior reactor operator (PEER) noted the initial load of the fuel assembly of approximately 1500 lbs, and failed to check the load again during travel. Both the refueling bridge operator and the senior reactor operator failed to note that the grapple and fuel assembly had rotated approximately 25 degrees out of position with the hoist box. At approximately 193 inches, the grapple actuator came in contact with the hoist box and began lifting the hoist box. The refueling machine weight gauge indicated an increase in weight to 2900 lbs.

The finding was of very low safety significance (GREEN) because it did not represent an actual event that upset plant stability or damaged fuel cladding. The licensee entered the violation in their corrective action program as Condition Report CR-WF3-2008-2423.

Attachment: Supplemental Information

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Anders, Manager, Plant Security
A. Buford, Engineer, System Engineering
K. Cook, Manager, Operations
J. Dorsey, Superintendent, Plant Security
C. Fugate, Assistant Manager, Operations
T. Gaudet, Director, Nuclear Safety Assurance
R. Gilmore, Manager, Corrective Action and Assessments
K. Gordon, Assistant Manager, Operations
M. Groome, Senior Lead Engineer, System Engineering
J. Kowalewski, General Manager, Plant Operations
J. Lewis, Manager, Emergency Preparedness
B. Lindsey, Manager, Outage
P. Mckenna, Technical Specialist, System Engineering
R. Murillo, Manager, Licensing
O. Pipkins, Senior Licensing Specialist, Licensing
R. Putnam, Manager, Programs and Components
B. Proctor, Manager, System Engineering
J. Ridgel, Manager, Quality Assurance
G. Scott, Engineer, Licensing
H. Thompson, Coordinator, Maintenance Projects
O. Tucker, Supervisor, System Engineering
D. Viener, Supervisor, Engineering
K. Walsh, Vice President of Operations
R. Williams, Senior Licensing Specialist, Licensing

NRC Personnel

S. Tingen, Senior Mechanical Engineer, Office of Nuclear Reactor Regulation

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000382/2008004-01	URI	Intersystem loss of coolant event (Section 1R15.1)
05000382/2008004-02	NCV	Inadequate pressure locking calculation (Section 1R15.1)
05000382/2008004-03	URI	Operability of safety injection Valves SI-405A/B (Section 1R15.2)
05000382/2008004-04	NCV	Failure to follow integrated EDG test procedure (Section 1R19)

Closed

05000382/2008004-02	NCV	Inadequate pressure locking calculation (Section 1R15.1)
05000382/2008004-04	NCV	Failure to follow integrated EDG test procedure (Section 1R19)
05000382/2008003-00	LER	Movement of fuel assembly in reactor vessel with fuel handling machine inoperable (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

Section 1R01: Adverse Weather Protection

NUMBER	TITLE	REVISION
OP-901-521	Severe Weather and Flooding	4
OP-901-314	Degraded Grid Conditions	0
ENS-DC-199	Off-Site Power Supply Design Requirements	3
ENS-PL-158	Switchyard and Transmission Interface Requirements	1
ENS-PL-159	Summer Reliability Plan	0

Section 1R04: Equipment Alignment

Condition Reports

CR-WF3-2005-0346	CR-WF3-2005-0833	CR-WF3-2005-1626	CR-WF3-2005-2642
CR-WF3-2005-3692	CR-WF3-2006-2115	CR-WF3-2006-2199	CR-WF3-2007-0705
CR-WF3-2008-0778	CR-WF3-2008-3257		

Procedures/Documents

NUMBER	TITLE	REVISION
OP-006-003	125V DC Electrical Distribution	301
OP-002-004	Chilled Water System	302

Work Orders

WR 53383	WR 93304	WR 134666	WR 130233
WR 93304	WO 86204	WO 90487	WO 90381

Section 1R05: Fire Protection

Procedures/Documents

NUMBER	TITLE	REVISION
UNT-005-013	Fire Protection Program	9
OP-009-004	Fire Protection	11
MM-007-010	Fire Extinguisher Inspection and Replacement	15
FP-001-015	Fire Protection System Impairments	17
Drwg. No. G-1375	Fire Protection - Reactor Auxiliary Building Plan EL. + 35.00’	
Drwg. No. G-1357	Fire Protection - Reactor Auxiliary Building Plan EL. - 35.00’	
Drwg. No. G-1379	Fire Protection - Reactor Auxiliary Building Plan EL. + 21.00’	
Drwg. No. G-1356	Fire Protection – Cooling Towers Plan	
Drwg. No. G-FP-0023	Fire Detection System Raceway and Equipment Layout - Reactor Auxiliary Bldg. EL + 35.00’	05/01/92

Section 1R07: Heat Sink Performance

Procedures/Documents

NUMBER	TITLE	REVISION
EPRI NP-7552	Heat Exchanger Performance Monitoring Guidelines	
PE-001-015	Administrative Procedure - Generic Letter 89-13 Heat Exchanger Test Basis	3
PE-001-014	Administrative Procedure - Heat Exchanger Performance Monitoring Program	2
PE-004-037	Administrative Procedure - Heat Exchanger	0

Section 1R11: Operator Requalification

Procedures/Documents

NUMBER	TITLE	REVISION
E-79	Simulator Scenario	1
OP-902-000	Standard Post Trip Actions	10
OP-901-202	Steam Generator Tube Leakage or High Activity	9
OP-901-212	Rapid Plant Power Reduction	3
OP-902-008	Safety Function Recovery Procedure	15
OP-901-510	Component Cooling Water System Malfunction	4
OP-902-000	Standard Post Trip Recovery	10
OP-902-001	Reactor Trip Recovery	11
OP-902-004	Excess Steam Demand Recovery	11

Section 1R12: Maintenance Effectiveness

Condition Reports

CR-WF3-2008-3566	CR-WF3-2008-3212	CR-WF3-2008-0704	CR-WF3-2007-1775
CR-WF3-2008-4182	CR-WF3-2008-2697	CR-WF3-2008-0703	CR-WF3-2007-0940
CR-WF3-2008-3563	CR-WF3-2008-1930	CR-WF3-2008-2897	CR-WF3-2007-4498
CR-WF3-2008-3527	CR-WF3-2008-1632	CR-WF3-2008-0129	CR-WF3-2007-4554
CR-WF3-2008-3437	CR-WF3-2008-1304	CR-WF3-2007-4554	CR-WF3-2007-4005
CR-WF3-2008-3380	CR-WF3-2008-1087	CR-WF3-2007-2509	CR-WF3-2007-3982
CR-WF3-2008-3232	CR-WF3-2008-0840	CR-WF3-2007-2507	CR-WF3-2007-3447
CR-WF3-2008-3229	CR-WF3-2008-0766	CR-WF3-2007-2112	CR-WF3-2007-3430

Procedures/Documents

NUMBER	TITLE	REVISION
DC-121	Maintenance Rule	1
NUMARC 93-01	Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	3
OP-903-014	Control Room Heating and Ventilation	301
OP-903-123	Control Room Envelope Pressure Test	300
MM-006-106	Plant Door Maintenance	303

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Condition Reports

CR-WF3-2007-4373	CR-WF3-2008-0975	CR-WF3-2008-1089	CR-WF3-2008-1436
CR-WF3-2008-4304			

Procedures/Documents

NUMBER	TITLE	REVISION
OI-037-000	Operations Risk Assessment Guideline	2
EN-WM-101	On-Line Work Management Process	1
OP-903-127	Reactor Trip Circuit Breaker Post Maintenance Test	3
OP-009-008	Safety Injection System	24
OP-002-003	Component Cooling Water System	305

Section 1R15: Operability Evaluations

Condition Reports

CR-WF3-1995-0477	CR-WF3-1995-0480	CR-WF3-1997-0852	CR-WF3-1998-1241
CR-WF3-1998-1243	CR-WF3-2000-1347	CR-WF3-2002-0468	CR-WF3-2002-0547
CR-WF3-2002-0678	CR-WF3-2003-2991	CR-WF3-20051362	CR-WF3-2007-3553
CR-WF3-2008-0848	CR-WF3-2008-0893	CR-WF3-2008-1671	CR-WF3-2008-2326
CR-WF3-2008-2611	CR-WF3-2008-2968	CR-WF3-2008-3000	CR-WF3-2008-3880
CR-WF3-2008-3976	CR-WF3-2008-4151	CR-WF3-2008-4161	CR-WF3-2008-4294

Procedures/Documents

NUMBER	TITLE	REVISION/DATE
EN-OP-104	Operability Determinations	3
OP-009-008	Safety Injection System	24
OP-008-003	Containment Cooling System	300
OP-002-003	Component Cooling Water System	305
OP-004-006	Core Protection Calculator System	301
OP-006-003	125V DC Electrical Distribution	301
	Letters from the licensee to the NRC concerning	Feb. 13, 1996

	Generic Letter 95-07	August 1, 1996
	Letters from the NRC to the licensee concerning Generic Letter 95-07	March 14, 1996 June 24, 1996
	NRC Generic Letter 95-07, "Pressure-Locking and Thermal-Binding of Safety-Related Power-Operated Gate Valves	August 17, 1995
	Waterford 3 Quality Assurance Program Manual	18
	Commonwealth Edison Company Pressure Locking Test Report	no date
	Electric Power Research Institute Report	no date
	TR-103229	
ECM 08-002	Susceptibility of SI 405A/B to Pressure Locking	0
ECM 91-076	SI 405A/B Actuator Thrust Calculation	3
	Electric Power Research Institute Report	Dec. 1999
	TR-114051	
EN-DC-126	Engineering Calculation Process	1
EN-OP-104	Operability Determination Process	3
EN-DC-149	Acceptance of Vendor Documents	2
	MPR Calculation - Pressure-Locking Evaluation of SI-405A/B	Sept. 11, 2008
	MPR Calculation - Pressure-Locking and Thermal Binding of EPRI MOVs	0
	Idaho National Engineering Laboratory Report INEL 96-00161 - Pressure-Locking Test Report	April, 1996
	Maintenance and Instruction Manual for Model 24RAH-A001 Shutdown Cooling Suction Isolation Valve Actuator	0

Section 1R18: Plant Modifications

Condition Reports

CR-WF3-2006-3584	CR-WF3-2006-4336	CR-WF3-2007-0136	CR-WF3-2007-0981
CR-WF3-2007-3575	CR-WF3-2008-0306	CR-WF3-2008-2291	CR-WF3-2008-2326

Procedures/Documents

NUMBER	TITLE	REVISION
EC-7920	Installation and Testing Requirement for SI-401A/B Motor Operator Modification	

Work Orders

00164192-03	00164193-03

Section 1R19: Postmaintenance Testing

Condition Reports

CR-WF3-2003-1408	CR-WF3-2003-1710	CR-WF3-2005-4730	CR-WF3-2007-0136
CR-WF3-2007-4125	CR-WF3-2008-0456	CR-WF3-2008-3598	CR-WF3-2008-3699
CR-WF3-2008-3700	CR-WF3-2008-3711		

Procedures/Documents

NUMBER	TITLE	REVISION
OP-903-053	Fire Protection System Pump Operability Test	14
OP-903-115	Train A Integrated Emergency Diesel Generator - Engineering Safety Features Test	9
EN-LI-119	Apparent Cause Evaluation Process	7
OP-009-008	Safety Injection System	24
OP-903-118	Primary Auxiliaries Quarterly IST Valve Tests	15
W3-DBD-038	Safety Related HVAC-Control Room	

Work Orders

51668276	111128-06	111128-07	114563-01
27977	81661	51644148	111128-08
131033	51644147		

Section 1R22: Surveillance Testing

Condition Reports

CR-WFR-20073967	CR-WFR-2007-3550	CR-WFR-2007-2866	CR-WFR-2008-1221
CR-WF3-2008-3624	CR-WF3-2008-3625		

Procedures/Documents

NUMBER	TITLE	REVISION
OP-903-120	Containment and Miscellaneous Systems Quarterly IST Valve Tests	07
OP-903-068	Emergency Diesel Generator and Subgroup Relay Operability Test	14
OP-903-030	Safety Injection Pump Operability Verification	15
OP-903-121	Safety System Quarterly IST Valve Test	08
OP-903-119	Secondary Auxiliary Quarterly IST Valve Test	09

Work Orders

51660827-01	51671498	51652078	51029559

Section 1EP6: Drill Evaluation

Procedures/Documents

NUMBER	TITLE	REVISION
OP-902-000	Standard Post Trip Actions	10
OP-902-002	Loss of Coolant Accident Recovery Procedure	12
EP-001-040	General Emergency	300
EP-001-001	Recognition and Classification of Emergency Conditions	22

Section 4OA1: Performance Indicator Verification

Performance Indicator Review Package 4th Quarter 2007
Performance Indicator Review Package 1st Quarter 2008
Performance Indicator Review Package 2nd Quarter 2008
Performance Indicator Review Package 3rd Quarter 2008

Section 4OA2: Problem Identification and Resolution

Condition Reports

CR-2007-1456	CR-2008-1931	CR-2008-3038	CR-2007-2717

Procedures/Documents

NUMBER	TITLE	REVISION
EN-LI-102	Corrective Action Process	12
OP-100-014	Technical Specification and Technical Requirements Compliance	301
EN-LI-119	Apparent Cause Evaluation Process	6
NEI White Paper	Treatment of Operational Leakage from ASME Class 2 and 3 Components	1
UNT-006-030	Administrative Control of External Corrosion	302

Section 4OA3: Event Followup

Procedures/Documents

NUMBER	TITLE	REVISION
OP-901-521	Severe Weather and Flooding	4
EPP-428	Emergency Facilities and Equipment Readiness	300
EP-003-060	Emergency Communication Guideline	5
EP-003-040	Emergency Equipment Inventory	5
EP-002-100	Technical Support Center (TSC) Activation, Operation, Deactivation	300

4OA5 Other

Procedures/Documents

NUMBER	TITLE	REVISION
PS-015-101	Security Patrols	302
08-033	Security Operations Bulletin	0

4OA7 Licensee Identified Violations

Condition Reports

CR-WF3-2008-2213	CR-WF3-2008-2414	CR-WF3-2008-2423	CR-WF3-2008-2427
CR-WR3-2008-2643	CR-WF3-2008-3196		

Procedures/Documents

NUMBER	TITLE	REVISION
RF-005-002	Refueling Equipment Operation	302
NE-001-005	Preparation, Control and Documentation of Fuel Movement	11
RF-005-001	Fuel Movement	301
RF-005-003	Refueling Machine and Operator Checkouts	302

LIST OF QUESTIONS PROVIDED TO THE LICENSEE IN WRITING

The NRC conducted an inspection of shutdown cooling valve malfunctions. The NRC utilized inspectors in the Arlington, Texas regional office and conducted on-site and in-office inspection. At times, the inspectors conducted conference calls from the Region IV office (Arlington, Texas) and communicated with the licensee (located in Killona, Louisiana) and the licensee's contractor MPR Associated Inc. (located in Alexandria, Virginia). To enhance communications concerning highly technical matters, the inspectors provided the following questions in writing to the licensee in advance of scheduled conference calls. The licensee forwarded some of these questions to their contractor, MPR Associates Inc. The licensee, and/or their contractor, answered most of the questions verbally. In a few instances documentation was provided, such as additional calculations, operability evaluations, and licensing documents. In some cases inspectors requested existing analysis to support licensee statements in licensing documents, but no analysis was available. A list of documents reviewed by the inspectors is included in this report. Findings associated with this inspection are documented in Section 1R15 of this report.

<u>Questions Concerning MPR Calculation 002156DHH-816, dated September 11, 2008</u>

- On Page A-1 of the MPR calculation, there are some caveats at the bottom of the page that specify certain conditions that might make the calculation's results erroneous. How do you know those conditions don't apply to this case? What is meant by high wedging loads? How much do loads and temperatures have to cycle in order to adversely affect the results?

- The calculation ran a few cases where the coefficient of disk-to-seat friction (μ_s) was 0.2, 0.35 or 0.5. Why would the total calculated thrust decrease for μ_s = 0.5, when compared to μ_s = 0.35? When the coefficient of friction increases, one would expect the calculated thrust to increase as well.

- Plant personnel have stated that they are following Electric Power Research Institute (EPRI) guidance for evaluating air operated valves. For these valves in these temperatures, EPRI recommends a μ_s of 0.52 to 0.55 (Page 5-32 of EPRI TR-107322, "Air-Operated Valve Evaluation Guide," dated May, 1999). What were the bases for the values of μ_s that were actually used in this calculation?

- In some places, the calculation uses the units of psi, when psia or psig would be appropriate. For example, the term Pbo = 2250 psi. Which units, psig or psia should be used?

- How was the MPR pressure locking calculation validated? Did it accurately predict when pressure locking would occur on test valves? If so, please provide the test data.

- The calculation appears to use a bounding value of bonnet pressure of 2250 psig. Is this correct?

- Are the results in the calculation applicable to both Valves SI-405A and SI-405B.

- For Valve 405A, we know that the valve did suffer a pressure locking event [Condition Report WF3-2008-02326, dated May 14, 2008]. Would this model have predicted this event?

- For Valve 405B, during the last shutdown, the valve unwedging force increased substantially over what was observed during diagnostic testing. Since unwedging forces do no typically change this much without an added component from pressure locking or thermal binding [or some other contributor like differential pressure], would this model have predicted the valve performance that was observed when attempting to open the valve for shutdown cooling operations?

- Typically in a valve that is pressure locked, as the valve stem tries to move outward, the liquid in the bonnet provides an equal and opposing force to the valve disc movement until enough fluid escapes (or gas in the bonnet is compressed) to permit valve disc movement. For example, in an absolutely solid bonnet (no gas) and a perfect seal between the top of the valve disc and the bonnet, bonnet pressure would increase to match the force that the actuator was attempting to apply. While no pressure-locked valve is this perfect, the same concept applies to some degree until enough fluid escapes (or gas is compressed) to allow valve disc movement. How does this model address this pressure locking behavior?

<u>Questions Concerning MPR Calculation 108-085-CSK-07, "Pressure-Locking and Thermal-binding Evaluation of EPRI MOVs [motor-operated valves]"</u>

- The MPR calculation used eight valves tested by EPRI during their pressure locking and thermal binding testing program. Were these the only valves tested in the pressure locking program? If not, why were valves excluded?

- The MPR calculation used two values for $\mu_{s.}$ One was based on an method recommended by EPRI. The other was called a "Best-Fit" μ_s. Were all values for calculated disk pull-out thrust contained in Table 2 derived the same (using the MPR method), with the only variables being the EPRI μ_s and the "Best-Fit" μ_s?

- What is the "Best-Fit" μ_s and why is it used? How is it determined?

- For three of the 8 valves tested by EPRI, the MPR method seemed to under-predict unseating thrust (Valves 24, 34A and 34B) when using the "Best-Fit" μ_s. In addition, of the 4 valves with bonnet pressures greater than 500 psig, the method seemed to under-predict thrust for three of these valves (same three valves). Finally, for all of the valves with bonnet pressures over 1000 psi, the model seemed to under-predict thrust (same three valves). Have you made adjustments to the model since validation to make it more conservative? NOTE: When using the EPRI μ_s, the calculated thrusts appeared conservative, when compared to actual measured pull-out thrusts.

- The Calculation makes reference to the "EPRI Gate Valve Model." Please provide a copy of the document that contains the model.

- Please provide an electronic version of either the pages that define the EPRI coefficient of friction calculation or the entire two volume set (whichever is easiest). I'll work through the parts that I need.

- Please provide an electronic version of the EPRI MOV testing report referred to in MPR's response to Question 1, below.

- Which coefficient of friction was used for the Waterford-3 calculation ("Best-Fit" or EPRI)?

- Does MPR have their own NRC Approved Quality Assurance Program or are they working under the licensee's Appendix B program?

- What is the uncertainty associated with the MPR calculation methodology?

- What other plants have used the MPR method?

<u>Questions for the licensee</u>.

- What were the tested seating thrusts for Valves SI-405A and SI-405B?

- What were the tested unseating thrusts for Valves SI-405A and SI-405B?

- What is the normal ambient containment temperature?

- From a picture of Valve 405B, it appears that there's no insulation on the top of Valve 405A and 405B. Is this correct?

- When bonnet temperatures were measured above 200°F, what was the bulk residual heat removal water temperature in the adjacent piping?

- Please provide a copy of all Waterford-3 (or applicable Entergy) procedures that govern the evaluation of third party (or contractor) calculations.

- Your June 17, 1999 supplemental response to the NRC concerning pressure locking and thermal binding stated, in part: "Normal ambient temperature variations are not sufficient to present concerns relative to thermally induced pressure locking " (page 2, 2nd paragraph). Did the plant perform an analysis that validated this statement? If so, please provide a copy of the analysis.

- The same response to the NRC stated, in part: "Following a SBLOCA [small break loss of coolant accident], it is expected that the temperature at . . . SI-405 will either remain constant or decrease prior to initiation of SDC [shutdown cooling]" (Page 2, 3rd paragraph). Did the plant perform an analysis that validated this statement? If so, please provide a copy of the analysis.

- The same response to the NRC stated, in part: "The decrease in RCS [reactor coolant system] temperature will cause the fluid in the bonnet to contract and depressurize thus eliminating any elevated pressure trapped in the bonnet" (Page 4, 2nd paragraph). Please provide any existing analysis which would validate this statement.

- The same response to the NRC stated, in part: "Operating history has shown that these valves have never failed to open during normal SDC operations." [page 4, last paragraph in the thermal binding section]. There appear to have been failures of these valves prior to your 1999 submittal - once during initiation of SDC and once during initiation of low temperature over-pressure protection. Why do you believe the information in your letter was accurate?

- Please provide plant computer generated data concerning the operation of Valve SI-405A, when it pressure locked on May 14, 2008.

- Concerning MPR Calculation 002156DHH-816, dated September 11, 2008, did your engineering staff review the calculation and approve it for use at Waterford-3? If so, please provide documentation associated with the review and approval.

NOTE: You had provided a CR number that contained a signature sheet. What I saw was a summary of conclusions which relied on the MPR calculation. I did not see a sheet that indicated that engineers had reviewed the calculation and found it technically adequate. If there was such an engineering review, I would still like to see a copy.

- Please provide an electronic version of the Waterford-3 "Quality Assurance Program Manual." This manual did not appear during an ADAMS search.

- Procedure EN-DC-126, Section 5.2[3] stipulates:

  "Calculations prepared in accordance with this procedure to support a reasonable expectation of operability <u>do not need to be design verified</u> [emphasis added]."

  The Waterford-3 "Quality Assurance Program Manual," Section B.3.d states:

  "Independent design verification is to be completed before design outputs are used by other organizations for design work and before they are used to support other activities such as procurement, manufacture, or construction. When this timing cannot be achieved, the unverified portion of the design is to be identified and controlled. In all cases, the <u>design verification is to be completed before relying on the item to perform its function</u> [emphasis added]."

  Waterford-3 appeared to use an MPR calculation for an operability evaluation without first performing a design verification. While this is consistent with one of your procedures, it appears inconsistent with your NRC approved "Quality Assurance Program Manual." Why do you believe this practice is acceptable?

- During a prior phone call, the licensee stated that the MPR method was similar to the ComEd method for calculating pressure locking.

  During an even earlier phone call, MPR provided reasons why with an increasing disk friction coefficient (from 0.35 to 0.5) the force required to move the stem would reduce.

  We reviewed the ComEd method to evaluate the calculation associated with the F-presslock term. From the ComEd paper, the term was represented as:

  $F_{pl} = [2\pi a(y_a/y_w)][\mu_s\cos\theta - \sin\theta] \times 2$

The first group of components calculates the load from pressure locking. There is no directional component from this load (it's not a vector), so there is no part of it that can help move the valve disc from the seat. Further, once calculated for a given set of pressures it's a constant. From this, with increasing disk to seat coefficients the load should also increase. Please explain why your method could predict a decreasing load (with an increasing seat coefficient of friction) while the ComEd method would predict an increasing load with an increasing disk coefficient of friction (assuming they are both the same).