UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

February 14, 2000

EA 2000-023

Charles M. Dugger, Vice President
Operations - Waterford 3
Entergy Operations, Inc.
P.O. Box B
Killona, Louisiana 70066

SUBJECT: NRC SPECIAL INSPECTION REPORT 50-382/99-25

Dear Mr. Dugger:

This refers to the inspection conducted on November 29 through December 3, 1999, at the Waterford Steam Electric Station, Unit 3, facility. The purpose of the inspection was to follow up on the reactor coolant system draindown event, which occurred on November 27, 1999. An exit interview to provide the results of this inspection was conducted on February 10, 2000. The enclosed report presents the results of this inspection.

Based on the results of this inspection, two apparent violations were identified and are being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The first apparent violation involved the failure to maintain Low Pressure Safety Injection Train B operable, as required by Technical Specification 3.5.2. The second apparent violation involved the failure to place a valve in the low pressure safety injection system in the correct position, as required by procedure. Since the NRC has not yet made a final enforcement decision, no Notice of Violation is being issued for these inspection findings. In addition, please be advised that the number and characterization of the apparent violations described in the enclosed inspection report may change as a result of further NRC review.

An open predecisional enforcement conference to discuss these apparent violations has been scheduled for March 20, 2000. The decision to hold a predecisional enforcement conference does not mean that the NRC has made a final determination that a violation has occurred or that enforcement action will be taken. This conference is being held to obtain information to enable the NRC to make an enforcement decision, such as a common understanding of the facts, root causes, missed opportunities to identify the apparent violation sooner, corrective actions, significance of the issues, and the need for lasting and effective corrective action. In addition, this is an opportunity for you to point out any errors in our inspection report and for you to provide any information concerning your perspectives on: (1) the severity of the apparent violations, (2) the application of the factors that the NRC considers when it determines the amount of a civil penalty that may be assessed in accordance with Section VI.B.2 of the

Enforcement Policy, and (3) any other application of the Enforcement Policy to this case, including the exercise of discretion in accordance with Section VII.

You will be advised by separate correspondence of the results of our deliberations on this matter. No response regarding these apparent violations is required at this time.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

Sincerely,

Elmo E. Collins for

Ken E. Brockman, Director
Division of Reactor Projects

Docket No.: 50-382
License No.: NPF-38

Enclosure:
NRC Inspection Report No.
50-382/99-25

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-382
License No.:	NPF-38
Report No.:	50-382/99-25
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	November 29 through December 3, 1999
Inspectors:	T. R. Farnholtz, Senior Resident Inspector R. E. Lantz, Reactor Engineer W. B. Jones, Senior Reactor Analyst
Approved By:	P. H. Harrell, Chief, Project Branch D

ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

Waterford Steam Electric Station, Unit 3
NRC Inspection Report 50-382/99-25

This special inspection reviewed aspects of operations, maintenance, engineering, and plant support activities related to the reactor coolant system draindown event, which occurred on November 27, 1999, while plant operators were attempting to place Train B of the shutdown cooling system in service.

Operations

- The failure to maintain the Low Pressure Safety Injection Train B system in an operable condition is an apparent violation of Technical Specification 3.5.2. An out-of-position valve resulted in a flow path that would divert a significant amount of flow from Low-Pressure Safety Injection Pump B to the refueling water storage pool instead of the reactor coolant system under accident conditions. No other emergency core cooling systems were affected by this condition. This issue is considered to be an apparent violation of Technical Specification 3.5.2 (Section O1.2).

- Plant automatic actions occurred, as expected, following the rapid loss of pressurizer level and reactor coolant system pressure. Operators in the control room responded appropriately to plant conditions and entered and properly executed the appropriate procedures for the indicated conditions (Section O1.3).

- A bottomed out position indicating a pin in the reach-rod operator for Valve SI-417B was identified as one barrier that was in place and failed to detect the mispositioned valve, which caused the reactor coolant system drain-down event. Three other additional missed opportunities to identify the mispositioned valve were also identified, which included procedural guidance to locally verify valve position for reach-rod operated valves and two computer mimic displays. The failure to place Valve SI-417B in the position specified in the applicable procedure is an apparent violation of Technical Specification 6.8.1 (Section O1.4).

Maintenance

- A review of the corrective maintenance history of Valve SI-417B did not reveal any concerns with the valve or reach-rod assembly. There was no evidence of recurring problems or ineffective maintenance on this component. A poor work practice was identified concerning the reuse of roll pins in reach-rod assemblies. Used pins were visually inspected and reinstalled if no abnormal wear or distortion was noted. The use of a new pin during reassembly would provide greater assurance that the pin would perform satisfactorily (Section M2.1).

- The 3-year preventive maintenance activity for the reach rod associated with Valve SI-417B was inadequate in that it did not include requirements to inspect any portion of the reach-rod assembly. Similar preventive maintenance activities were performed on other reach-rod operators associated with safety-related valves based on

the necessity for their usage. This resulted in a significant number of reach-rod operators used on safety-related valves that did not receive regularly scheduled preventive maintenance or inspection (Section M2.2).

- The licensee's initial efforts to determine the extent of the condition of other reach rods in the plant following the failure of the reach rod for Valve SI-417B was inadequate. An inspection was performed on approximately 79 percent of the safety- and nonsafety-related reach-rod assemblies. However, the inspections were incomplete in that two previously unknown pins for safety-related operators similar to that used on Valve SI-417B was subsequently identified. Inspections of these additional pins resulted in numerous discrepancies. The general condition of reach-rod operators in the plant was considered marginally adequate and reflected the licensee's lack of adequate preventive maintenance and inspection programs for these components (Section M2.3).

<u>Engineering</u>

- The NRC staff concluded, from a qualitative assessment, that the event was of moderate risk significance. This finding was based on the required operator actions to isolate the shutdown cooling bypass to the refueling water storage pool and the initiation of reactor coolant system inventory makeup. The risk was minimized through the redundancy available to isolate the bypass leakage and the availability of both high-pressure safety injection trains and Low-Pressure Safety Injection Pump A (Section E1.1).

- The licensee's efforts to assess the effect of the reactor coolant system drain-down event on plant equipment and to establish appropriate conclusions and recommendations was adequate. The conclusions and recommendations were based on appropriate data obtained from various sources. The root cause of the event was identified as a roll pin falling out of a reach-rod operator linkage due to the impact forces that it had been subjected to over many cycles of the valve. The licensee's actions and recommendations for short- and long-term corrective actions were adequate (Section 1.2).

- The licensee's actions to restore Low Pressure Safety Injection Train B to an operable status were adequate. Extensive walkdowns, inspections, evaluations, and testing were performed to ensure the integrity of the individual components and the system as a whole (Section E2.1).

- The licensee's methods for quantifying potential voiding in the reactor coolant system were reasonable. The resulting numbers indicated that a most probable value of approximately 97 to 110 gallons (13 to 15 cubic feet) of voiding may have occurred. The volume of voiding required to inhibit establishing single-phase, natural circulation conditions is approximately 5700 gallons (Section E2.2).

Plant Support

- The control room staff properly implemented the facility Emergency Plan in response to the reactor coolant system draindown. The Alert declaration was appropriate and the required notifications to state and local officials, as well as the NRC, were appropriate and timely (Section P3.1).

Report Details

Summary of Plant Status

At the beginning of this inspection period, the plant was shut down in Mode 5 to perform repairs to a steam line upstream of a main steam isolation valve. Repairs were completed and the plant was started up on December 1, 1999. The main generator was connected to the electrical grid on December 2 and power was increased to approximately 100 percent and remained at that level for the remainder of this inspection period.

I. Operations

O1 Conduct of Operations (71707)

O1.1 Introduction

Summary of Event

On November 27, 1999, at 4:47 a.m., the plant was in Mode 4 with two reactor coolant system (RCS) pumps in operation. The licensee was making preparations to place Train B of the shutdown cooling system in service and proceed to Mode 5 to allow repairs to a steam line upstream of Main Steam Isolation Valve 2. Plant operators were at the point in the procedure where the last motor-operated valve (SI-407B) was to be opened to complete the lineup and the low-pressure safety injection (LPSI) pump was to be started.

Immediately after a control room operator began to open Valve SI-407B, RCS pressure and pressurizer level decreased rapidly. At the same time, an auxiliary operator stationed in the vicinity of the LPSI Train B pump heard a thud and observed the LPSI Train B pump rotating, even though the pump had not been energized. Operators secured the running RCS pumps when RCS pressure dropped below the minimum requirements for net positive suction head for these pumps. Also, the pressurizer heaters automatically tripped on low pressurizer level (to protect the heaters) and a second charging pump automatically started.

Control room operators placed the switch for Valve SI-407B to close. The valve stopped in midstroke in the open direction, reversed direction, and began to go closed. The valve stroked to the full closed position. RCS pressure began to rise and the auxiliary operator near the LPSI Train B pump heard another thud and observed the LPSI pump slow and stop. Pressurizer level came back on scale and the operators started High-Pressure Safety Injection (HPSI) Pump A to increase pressurizer level to approximately 35 percent cold calibration (cal).

RCS pressure prior to the event was 353 psia, RCS temperature was 291°F, pressurizer level was 35.77 percent cold cal, and refueling water storage pool (RWSP) level was 85.3 percent. During the event, the lowest pressurizer pressure instrument indicated 105 psia, the highest core exit thermocouple indicated 308°F, the cold cal pressurizer level instrument indicated that the pressurizer had drained to below the indicating range, and the RWSP level indicated approximately 86 percent.

Following the event, the shift superintendent determined that activation of the Technical Support Center was warranted given the unknown nature of the cause of the event. An Alert was declared at 5:33 a.m. (CST) on November 27.

The licensee determined that an out-of-position valve (SI-417B) was the cause of this event. This was a normally closed valve that was found to be in the open position. This valve was last operated on November 19 for the purpose of establishing recirculation flow in the RWSP. This reach-rod operated valve was thought to have been repositioned to the closed position later that same day, but the reach rod had become disconnected because a pin in a coupling had fallen out. With Valve SI-417B open, a flow path was available from the RCS, through the LPSI Train B pump, to the RWSP.

System Description

The LPSI Train B system, operating in the shutdown cooling mode, provides forced flow through the reactor core and the shutdown cooling system heat exchanger to remove decay heat. The system consists of a motor-operated pump, a shell and tube heat exchanger, and the necessary piping and valves to form a loop from the RCS, through LPSI Train B pump, Train B shutdown cooling heat exchanger, and back to the RCS. An additional flow path from the discharge side of the LPSI Train B pump is through 6-inch Valve SI-417B to the RWSP. The primary purpose of this flow path is to allow use of the LPSI system to recirculate the contents of the RWSP to maintain a uniform boron concentration.

O1.2 Technical Specification (TS) Requirements Related to the Operation of the LPSI System

a. Inspection Scope (93702)

The inspectors evaluated the circumstances and facts surrounding the inadvertent transfer of reactor coolant from the RCS to the RWSP through the LPSI Train B system. The inspectors reviewed the sequence of events and associated documentation and conducted interviews with selected plant personnel.

b. Observations and Findings

The following is a sequence of events associated with this event (all times are CST):

November 19, 1999

4:27 p.m.	Valve SI-417B opened for RWSP recirculation
5:15 p.m.	Placed RWSP on recirculation using LPSI Pump B
10:35 p.m.	RWSP recirculation secured. Valve SI-417B is shut (the valve failed to close because the reach rod failed to properly function)

<u>November 26, 1999</u>

12:55 p.m.	Commenced a plant shutdown to repair a steam leak from unisolable drain line upstream of Main Steam Isolation Valve 2
3:11 p.m.	Plant entered Mode 3
10:59 p.m.	Plant entered Mode 4

<u>November 27, 1999</u>

4:41:00 a.m.	Shutdown cooling system Train B is aligned in standby in accordance with Procedure OP-009-005, "Shutdown Cooling System," Revision 14
4:47:00 a.m.	Plant conditions are as follows: (1) wide-range Channel A pressurizer pressure 353 psia, (2) representative core exit thermocouple 293°F, (3) RCS hot leg temperature 291°F, (4) pressurizer level 35.77 percent cold cal, and (5) RWSP average level 85.3 percent
4:47:36 a.m.	Valve SI-407B begins to open. RCS pressure and pressurizer level begin to drop, RWSP level begins to rise. Auxiliary operator in Safeguards Room B reports hearing a thud and observes LPSI Pump B rotating.
4:48:00 a.m.	Control room operator places Valve SI-407B switch to close
4:48:02 a.m.	Pressurizer heaters automatically trip at 28 percent pressurizer level
4:48:03 a.m.	Charging Pump B automatically started (two charging pumps now running)
4:48:04 a.m.	Letdown flow at minimum
4:48:34 a.m.	Pressurizer level at 5 percent cold cal
4:48:47 a.m.	Valve SI-407B indicates closed. Pressurizer level indicates 1.9 percent cold cal (bottom of indicating range). RWSP average level 86 percent and rising
4:48:53 a.m.	Reactor Coolant Pump 2B secured
4:48:55 a.m.	Reactor Coolant Pump 1B secured

4:49:20 a.m.	Wide-range Channel A pressurizer pressure lowest pressure indicates 105 psia. Pressure begins to rise
4:49:00 a.m.	Auxiliary operator in Safeguards Room B reports hearing another thud and observes LPSI Pump B slow and stop
4:52:21 a.m.	Highest representative core exit thermocouple 308°F, wide-range Channel A pressurizer pressure 121 psia
4:53:00 a.m.	RWSP average level peaks at 86.5 percent and begins to decrease
4:57:31 a.m.	Pressurizer level at 5 percent cold cal and increasing
5:04:00 a.m.	RWSP average level 86.22 percent and slowly decreasing
5:07:00 a.m.	HPSI Pump A started to make up to the RCS
5:09:00 a.m.	Charging pump suction shifted to RWSP. HPSI cold leg injection at 100 gpm into each of the RCS legs
5:09:12 a.m.	RWSP average level 86.10 percent
5:11:00 a.m.	Hot leg injection initiated from HPSI Pump A at 50 gpm
5:13:01 a.m.	Pressurizer heaters reenergize
5:15:00 a.m.	Secured hot leg injection
5:16:00 a.m.	Secured cold leg injection. Pressurizer level at 35 percent cold cal
5:21:00 a.m.	Core exit thermocouple 296°F and decreasing
5:33:00 a.m.	Plant entered Alert condition
11:06:00 a.m.	Placed shutdown cooling system Train A in service
12:26:00 p.m.	Plant exited Alert condition
12:35:00 p.m.	Plant entered Mode 5

The inspectors reviewed this sequence of events and determined that an apparent violation of TS requirements occurred between November 19 at 4:27 p.m. and November 26 at 10:59 p.m. TS 3.5.2 requires that two independent emergency core cooling system (ECCS) subsystems be operable while in Modes 1, 2, and 3. Included in this requirement is an operable LPSI pump and an operable flow path. With

Valve SI-417B in the open position, a flow path existed from the discharge side of the LPSI pump back to the RWSP. With this abnormal lineup, a significant amount of flow would be diverted to the RWSP instead of into the RCS under accident conditions, thus rendering LPSI Train B inoperable. This condition existed for approximately 7 days, which exceeded the TS-allowable outage time for LPSI Train B of 72 hours.

The inspectors reviewed drawings of other ECCS systems to determine if any other systems would be affected by open Valve SI-417B. This review included LPSI Train A, HPSI Trains A and B, and containment spray Trains A and B. The inspectors noted that the operability of these systems would not be affected with Valve SI-417B in the open position.

The failure to maintain LPSI Train B in an operable condition is an apparent violation of TS 3.5.2. This issue is an apparent violation of TS 3.5.2 (50-382/9925-01).

c. Conclusions

The failure to maintain Low-Pressure Safety Injection System Train B in an operable condition is an apparent violation of Technical Specification 3.5.2. An out-of-position valve resulted in a flow path that would divert a significant amount of the flow from Low-Pressure Safety Injection Pump B to the refueling water storage pool instead of the RCS under accident conditions. No other ECCS was affected by this condition. This issue is considered an apparent violation of Technical Specification 3.5.2.

O1.3 Plant and Operator Response

a. Inspection Scope (93702)

The inspectors reviewed recorded plant data, operator logs, plant procedures, system descriptions, training material, and event debrief statements to assess the automatic plant response and immediate and follow-on actions of the operators to the RCS drain-down event.

b. Observations and Findings

On November 19, 1999, following completion of a successful recirculation of the RWSP, Valve SI-417B, “LPSI B recirc isolation to RWSP,” was left open following completion of the restoration valve lineup. This condition was unknown until November 27, following the drain-down event.

On November 27, operators began the shutdown cooling valve lineup to initiate Train B of shutdown cooling to enter Mode 5 and had reached the final valve (SI-407B) in the lineup prior to starting LPSI Pump B. Valve SI-407B is a 6-inch, motor-operated, gate valve that is controlled remotely from a control board in the control room. It has a seal-in feature, such that the operator positions the control switch to open, then releases the switch, which spring-returns to neutral. The valve then continues to stroke toward the open direction, which requires approximately 63 seconds. The control room operator placed the control switch to open and observed the valve position indication change, indicating the valve was opening. Within a few seconds, pressurizer level began dropping, with a corresponding drop in RCS pressure. At the expected low pressurizer level setpoints, all pressurizer heaters de-energized and the second charging pump started automatically. Operators verified these expected automatic actions.

Approximately 35 seconds after placing the control switch to open, and following reports from the local operators of pump rotation and banging noises in LPSI Train B, the control room operator placed the control switch for Valve SI-407B to close. The valve stopped opening, immediately started to close, and fully closed approximately 35 seconds later. The operators also tripped the two running RCS pumps as an immediate action due to the reduction in RCS pressure to below the minimum pressure for RCS pump operation. Although a caution was contained in Procedure OP-100-009, “Control of Valves and Breakers,” which stated that the direction of nonthrottling, motor-operated valves should not be reversed in midtravel due to the potential for tripping the breaker or circuit damage, the inspectors agreed that the actions taken by the control room operators were an appropriate immediate response given the unexpected plant indications.

Based on indications of rapidly lowering pressurizer level and pressure, the control room operators entered Abnormal Operating Procedure OP-901-111, “Reactor Coolant System Leak.” The first step of this procedure directed a transition to Procedure OP-901-131, “Shutdown Cooling Malfunction,” which was then entered by the operators. Procedure OP-901-131 was appropriately referenced and completed, which directed concurrent reference to Procedure OP-902-000, “Emergency Entry Procedure.” This procedure directed the start of HPSI Pump Train A to restore pressurizer level and restoration of RCS cooling by steaming an available steam generator. The operators completed the actions, as directed, and restored cooling with natural circulation and steaming to the main condenser.

c. Conclusions

Plant automatic actions occurred, as expected, following the rapid loss of pressurizer level and RCS pressure. Operators in the control room responded appropriately to plant conditions and entered and properly executed the appropriate procedures for the indicated conditions.

O1.4 Failed Barriers

a. Inspection Scope (93702)

The inspectors reviewed the circumstances and events leading to the RCS drain-down to identify any barriers that were in place that could have but failed to prevent the drain-down event from occurring. The inspectors also reviewed the plant activities during the 8-day period, that Valve SI-417B was open and out of position, to identify missed opportunities to discover the out-of-position valve.

b. Observations and Findings

On November 19, Valve SI-417B was unlocked and opened in preparation for placing the RWSP on recirculation. This was being done to ensure the boron concentration in the RWSP was not stratified prior to filling a safety injection tank (SIT). After the RWSP had been recirculated in accordance with the procedure, two plant operators were sent to Valve SI-417B to close and lock the valve as part of the restoration valve lineup for LPSI Train B.

The inspectors interviewed the two operators, one of which was the initial valve positioner. The second served as a peer-check and valve position verifier. Both of these operators had been auxiliary operators in the plant for greater than 5 years. The initial valve positioner stated that, prior to operation of the valve, he independently observed the reach-rod linkage for Valve SI-417B from the floor of the room in which it is located. The valve position verifier had not observed the linkage, but did verify that the initial valve positioner had done so by questioning him prior to valve manipulation. Both operators went together to the valve gallery, where the reach-rod handwheel operator for Valve SI-417B is located. Following a peer check, the initial valve positioner unlocked and rotated the handwheel for Valve SI-417B to the indicated fully closed position. The initial valve positioner stated that he did not notice any unusual movement of the handwheel. The valve position verifier then concurred that the valve indicated fully closed, and then the valve handwheel was locked and verified. Neither operator returned to the location of the valve after the valve had been manipulated.

The local indication of valve position for Valve SI-417B, as well as many other reach-rod operated valves, consists of a pin that moves up and down in a slotted steel plate, with open and closed position marks. On November 19, when the operator attempted to close Valve SI-417B, the indicating pin did travel from the open to the closed indication mark; however, the pin traveled fully to the end of the slotted plate, which restricted any further movement of the pin. The operators, as well as training and operations management, failed to recognize that this could have indicated a partially open valve or broken reach-rod assembly, which the inspectors considered as the first missed opportunity to identify the mispositioned valve.

Procedure OP-100-001, “Duties and Responsibilities of Operators on Duty,” step 5.13.5, stated, in part, that operators shall comply with both the content and <u>intent</u> of approved procedures. Procedure OP-100-009, “Control of Valves and Breakers,” step 5.3.9, stated, in part, that the checking of a valve normally operated by reach rod or other manual remote operator, should be done locally at the valve <u>when</u> possible. The valve position may be checked at the remote position indication if a local check would present a safety or radiation hazard or otherwise conflict with ALARA if the valve is not accessible for other reasons.

The valve was accessible; however, a local position check was not completed by the operators. One of the operators had done a visual-only inspection of the reach-rod mechanism for indications of a disconnected linkage or other gross failure. The inspectors interviewed plant operations management, as well as training instructors and the operations training supervisor, and verified that the method of ensuring proper valve operation for Valve SI-417B did meet minimum facility expectations and was consistent with current training. The inspectors considered the facility expectation for implementation of the procedural guidance for locally verifying valve position as inadequate, and it was the second missed opportunity to identify the mispositioned valve.

The licensee issued a memorandum to all Operations Department personnel, on November 30, 1999, to revise management expectations for verification of valve position for reach-rod-operated valves. The expectations clarified that a positive method of position verification must be conducted following operation of the reach-rod-operated valve, which preferably was a check locally at the valve itself, where possible. The memorandum also discussed when local verification would not be considered viable and presented acceptable alternate methods of verification, which included computer point data and system response verification. The inspectors considered the expectation clarification as an adequate immediate corrective action to the valve verification error.

On November 23, approximately 4 days after recirculation of the RWSP, operators performed the procedure to fill a SIT. Although not required in a procedure, a plant computer mimic (SIT 5) could have been displayed, as is typical practice when filling a SIT. This mimic would have shown that Valve SI-417B was open. The inspectors could not confirm if the mimic had been displayed; however, it was considered as the third missed opportunity to identify the mispositioned valve.

On November 27, immediately before the drain-down event, the operators could have displayed the plant computer shutdown cooling mimic, which contained the position display for Valve SI-417B. When questioned about the shutdown cooling mimic, operators did not recall from memory that Valve SI-417B was on the mimic. Display of the mimic was not required by procedure; however, typical practice was for the operator to display this mimic immediately before starting the LPSI pump to provide additional indications of pump start and running parameters, such as system flow and pump amperage. The next step in the procedure for starting shutdown cooling following opening of Valve SI-407B was to start LPSI Pump B. If an operator had displayed the mimic prior to opening Valve SI-407B, mispositioned Valve SI-417B may have been observed. This was considered as the fourth missed opportunity to identify the mispositioned valve.

Procedure OP-009-008, "Safety Injection System," Revision 15, Section 8.7, required, in part, that Valve SI-417B be shut after completion of the evolution to recirculate water in the RWSP using the LPSI pump. Attachment 11.1 of this same procedure required, in part, that the required position for Valve SI-417B is locked closed. The licensee failed to perform these procedures, as written, in that Valve SI-417B was not placed in the locked closed position after the RWSP had been recirculated. This is an apparent violation of TS 6.8.1 (50-382/9925-02).

c. Conclusions

A bottomed out position indicating pin in the reach-rod operator for Valve SI-417B was identified as one barrier that was in place and failed to detect the mispositioned valve, which caused the drain-down event. Three other additional missed opportunities to identify the mispositioned valve were also identified, which included procedural guidance to locally verify valve position for reach-rod operated valves, and two computer mimic displays. The failure to place Valve SI-417B in the position specified in the applicable procedure is an apparent violation of Technical Specification 6.8.1.

II. Maintenance

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Corrective Maintenance Performed on Valve SI-417B

a. Inspection Scope (93702)

The inspectors reviewed the corrective maintenance that had been performed on Valve SI-417B to determine if there was a history of problems or conditions that would indicate a potential cause for the reach rod failure.

b. Observations and Findings

Following the event of November 27, the inspectors requested a complete history of maintenance tasks that had been performed on Valve SI-417B. This valve was found to be in an incorrect position due to a broken reach-rod operator.

Since 1987, there were a total of seven maintenance action items written against Valve SI-417B. The majority of these items were for repairs to the remote position indication equipment or for packing leaks. One item from 1991 indicated that some work was done on the reach-rod assembly to repair a broken linkage. In addition, at least two items appeared to involve the removal and reinstallation of the reach-rod assembly to complete the work. These tasks involved repairs for a body-to-bonnet leak and a worn upper yoke bearing/bushing.

The review of the corrective maintenance history of this valve did not reveal any concerns with the valve itself or the reach-rod operating mechanism. There was no evidence of recurring problems or ineffective maintenance on this component.

During interviews with maintenance personnel, the inspectors were informed that it was not routine practice to replace roll pins with a new one when they were removed for maintenance. The maintenance technicians inspected the pins and reinstalled them (not necessarily in the same location) if there were no signs of abnormal wear or distortion during reassembly of the reach rod. The inspectors considered this to be a poor work practice since the effectiveness of a roll pin is heavily dependent upon the spring forces that the pin will exert on the hole in which it is installed. The effectiveness of a used roll pin cannot be adequately determined by visual inspection. The use of a new roll pin each time an old one is removed would provide greater assurance that the pin would perform satisfactorily.

c. Conclusions

A review of the corrective maintenance history of Valve SI-417B did not reveal any concerns with the valve or reach-rod assembly. There was no evidence of recurring problems or ineffective maintenance on this component. A poor work practice was identified concerning the reuse of roll pins in reach-rod assemblies. Used pins were visually inspected and reinstalled if no abnormal wear or distortion was noted. The use of a new pin during reassembly would provide greater assurance that the pin would perform satisfactorily.

M2.2 Preventive Maintenance (PM) Program for Reach-Rod Operators

a. Inspection Scope (93702)

The inspectors reviewed the licensee's PM program for the reach-rod operator associated with Valve SI-417B, in particular, and reach rods associated with safety-related valves in general, to determine if it was adequate to ensure proper operation of these components.

b. Observations and Findings

The licensee's PM program for reach-rod operators associated with safety-related valves included the reach-rod for Valve SI-417B. The inspectors reviewed the PM task for this reach rod to determine if it was adequate to identify potential problems of the type that resulted in the RCS drain-down event.

It was determined that a roll pin had fallen out of the reach-rod operator, which caused the valve to become disconnected from the remote handwheel. The PM task performed on Valve SI-417B involved lubricating the reach-rod assembly and cycling the valve. The task did not require inspection of any portion of the reach rod. Because of this, there was no evidence that the roll pins had been inspected on the assembly. The PM task was on a 3-year frequency and was last performed on Valve SI-417B on August 18, 1998. The inspectors considered the lack of inspection requirements in the reach-rod PM program to be inadequate to ensure proper operation of these assemblies. The licensee stated that the PM program would be reviewed and changes made, as appropriate. These changes will include more detailed inspection criteria and will include all safety-related valves.

The inspectors also examined the licensee's overall PM program for all reach-rod operators. There are a total of 230 reach-rod operated valves, both safety- and nonsafety-related. Of these, 81 were safety-related and 149 nonsafety-related. No PM tasks existed for the reach rods associated with nonsafety-related valves. Of the

81 reach rods for safety-related valves, 22 were included under a similar PM task as that described for Valve SI-417B above. The valves included in this program were selected based on a necessity to operate them during the performance of an off-normal procedure, emergency operating procedure, or inservice test procedure. The inspectors considered this to be an adequate selection method, although this left 59 reach-rod operators associated with safety-related valves with no regularly scheduled maintenance or inspection activities.

c. Conclusions

The 3-year preventive maintenance activity for the reach rod associated with Valve SI-417B was inadequate in that it did not include requirements to inspect any portion of the reach-rod assembly. Similar preventive maintenance activities were performed on other reach-rod operators associated with safety-related valves based on the necessity for their usage. This resulted in a significant number of reach-rod operators used on safety-related valves that did not receive regularly scheduled preventive maintenance or inspection.

M2.3 Inspection of Reach-Rod Operators

a. Inspection Scope (93702)

The inspectors reviewed the licensee's efforts and performed independent inspections to determine the extent of the condition identified as one of the causes of the RCS drain-down event.

b. Observations and Findings

To determine the extent of the condition of the reach-rod operator problem identified on the reach-rod assembly for Valve SI-417B, the licensee performed inspections of a number of additional reach-rod operating mechanisms associated with both safety- and nonsafety-related valves in the plant. In addition, the inspectors examined a number of reach-rod assemblies to independently assess the condition of these components.

Following the RCS drain-down event on November 27, the licensee performed inspections of 58 of the 81 reach-rod operators associated with safety-related valves to determine their condition. This represented approximately 72 percent. In addition to replacing all five roll pins in the reach-rod operator associated with Valve SI-417B to return it to operable condition, the licensee identified two other reach-rod operators associated with safety-related valves with minor discrepancies. These were Valves CVC-150C (pin was deformed on one end but still tight) and CVC-152C (pin was worn to an oval shape but still tight).

On December 9 and 10, the licensee performed additional inspections of reach rods for safety-related valves that were of a similar design as Valve SI-417B. It was discovered that previously unknown pins were located in the pedestal area of these operators. There were a total of 15 reach rods of similar design for Train A and 19 for Train B valves. Of these, 13 were inspected on Train A and 17 were inspected on Train B. The remaining four were in the boron management system (Valves BM-233A, B, C, and D) and located in high radiation areas. Two of these operators were subsequently inspected and determined to be of a different design, which did not use pins in the pedestal area. Based on a review of applicable drawings, the licensee concluded that the remaining two boron management valve operators did not have pins in this area. The results of these inspections were as follows:

Train A Valves

SI-202A	Top pin hole out-of-round
SI-212A	Bottom pin out 1/8 inch
BM-232A	Bottom pin short 3/4 inch on one side and 1/4 inch on the other side
SI-203A	Top pin hole out-of-round
SI-417A	Bottom pin hole out-of-round
SI-208A	Top pin out 1/8 inch
CS-117A	Top pin loose
SI-410A	Top pin 1/4 inch short

Train B Valves

CS-117B	Bottom pin out 1/8 inch
CS-101B	Top pin out 3/8 inch, bottom pin out 1/8 inch
CVC-203	Bottom pin out 1/4 inch
SI-418B	Bottom pin out 1/8 inch, top pin loose
SI-208B	Bottom pin may be loose
SI-410B	Top pin out 1 1/4 inches
SI-212B	Top pin out 1/8 inch and loose
SI-202B	Bottom pin 1/8 inch short
SI-203B	Top pin hole out-of-round
CVC-1654B	Bottom pin hole out-of-round

Based on these results, the inspectors considered the licensee's initial efforts to determine the extent of the conditions of reach rods to have been inadequate. The initial representation of the identified discrepancies was not accurate or complete. Because of this, the true condition of the reach-rod operators used on safety-related valves could not be accurately assessed based on the results of the licensee's initial inspections.

The licensee performed operability assessments of these valves following identification of these discrepancies. The assessments determined that all affected valves remained operable with the discrepancies noted. The inspectors reviewed these operability assessments and identified no concerns. The identified discrepancies on the Trains A and B pedestal reach-rod assemblies were corrected in the short term by reinserting pins that were identified as tight but protruding, replacing loose pins, replacing pins identified as too short, and replacing the pin for Valve SI-410B. In addition, maintenance action items were generated to repair all identified discrepancies. The inspectors considered these measures adequate for short-term corrective action.

In addition to the reach rods used on safety-related valves, the licensee inspected 124 of the 149 nonsafety-related, reach-rod mechanisms. This represented approximately 83 percent. During these inspections, the licensee identified one discrepancy on the reach rod for Valve NG-230B (the pin was backed out about 3/8-inch but still tight).

The design used for the reach-rod operators associated with both the safety- and nonsafety-related valves were essentially the same. The same components were used during construction, including the roll pins. Therefore, the inspectors considered the overall sample size (182 out of 230 inspected or 79 percent) to be adequate to ensure a reasonable assessment of the condition of reach-rod operators. However, the licensee's initial inspection efforts did not provide all relevant information and was not adequate to give a true picture of the overall condition of these components. Subsequent inspection allowed a more accurate assessment. The majority of the discrepancies noted were relatively minor in nature and did not render any associated reach-rod assembly inoperable. The inspectors considered the overall conditions of the reach-rod operators to have been marginally adequate. The discrepancies also reflected the licensee's lack of adequate preventive maintenance or inspection of these components.

c. Conclusions

The licensee's initial efforts to determine the extent of the failed reach rod for Valve SI-417B was inadequate. An inspection was performed on approximately 79 percent of the safety- and nonsafety-related, reach-rod assemblies. However, the inspections were incomplete in that two previously unknown pins for safety-related operators similar to that used on Valve SI-417B was subsequently identified. Inspections of these additional pins resulted in numerous discrepancies. The general condition of reach-rod operators in the plant was considered marginally adequate and reflected the licensee's lack of adequate PM and inspection programs for these components.

III. Engineering

E1 Conduct of Engineering (37551)

E1.1 Risk Assessment

a. Inspection Scope (93702)

The NRC staff reviewed the risk associated with the LPSI Valve SI-417B being open for 8 days and the conditional core damage probability associated with the loss of RCS inventory during alignment for shutdown cooling.

b. Observations and Findings

The at-power risk significance was assessed for the 8 days Valve SI-417B was in the open position. The LPSI train has a risk achievement worth of approximately 2. The LPSI system is important in the plant success criteria for large break loss-of-coolant accidents (low frequency event of approximately 5E-5/yr) and is also important for establishing shutdown cooling following a steam generator tube rupture. The licensee performed an analysis that estimated that the normal flow from LPSI Train B to the reactor vessel, through Valve SI-417B, would be reduced by approximately 800 gpm, but would still be sufficient to meet the large break loss-of-coolant accident success criteria. The LPSI pumps are automatically tripped during a recirculation actuation signal and require operator action to reinitiate. The LPSI system is not required for short- or long-term recirculation. The resulting risk from a steam generator tube rupture would be the same as the initiating probability of a steam generator tube rupture for the 8-day period and the conditional risk from placing LPSI Train B in the shutdown cooling mode of operation. The NRC staff determined that the conditional core damage probability during the 8-day period that Valve SI-417B was open was low. This finding was consistent with the licensee’s evaluation.

The NRC staff considered the overall risk significance of the loss of RCS inventory during the initial alignment of LPSI Train B for shutdown cooling. The event risk significance was evaluated for consideration in the NRC followup and overall event significance. The event was initially assessed in a qualitative manner during the period the licensee was in an Alert. The staff considered three elements in the initial recommendation that the core damage probability of the event was sufficiently low to respond to the event with the augmented NRC resident inspector staff on site and to provide a followup special inspection. The three elements considered were the operators' ability to identify the condition and take actions to mitigate the initiator, redundancy of systems available to mitigate the event, and the potential for a common-cause failure resulting from the event that could have prevented or complicated the operators' recovery actions. The similarity of this event to other shutdown cooling events, which resulted in a loss of RCS inventory at other nuclear plants, was considered to bound the risk significance. Throughout this event, the risk importance was driven by the operator actions to isolate the leak, establish safety injection for inventory control, and either establish natural circulation or initiate the second train of shutdown cooling.

The initiating event was the loss of RCS inventory, which occurred when LPSI Valve SI-407B was opened to establish the alignment for shutdown cooling. The immediate plant response to the opening of Valve SI-407B led to the response by the operators to close Valve SI-407B before it had fully stroked open. The response to close the valve was assigned a low probability of failure. The probability of the valve tripping as a result of the valve stopping during its opening cycle and reversing was considered the dominant factor in failing to immediately isolate LPSI Train B. Subsequent actions to isolate the shutdown cooling suction line from the RCS were given a low failure probability. This was based on Valve SI-407B tripping and a minimum of 10 minutes for the operators to close, from the control room, one of two other motor-operated valves in the same line. The operator action to initiate HPSI on decreasing RCS level was an immediate action and given a low probability of failure.

The NRC staff considered that the two HPSI pumps and the LPSI Train A pump were available to immediately re-establish RCS inventory. The SIT isolation valves could also have been opened to provide for reflooding of the RCS within 10 minutes of the initiating event. The NRC staff assigned a high probability of failure of restoring the RCS inventory using the SITs.

Potential for common-cause failure of the two HPSI trains and the remaining LPSI train was specifically considered. The NRC staff did not identify a failure mechanism that would have resulted in the common-cause failure of these systems. This included a review of the RWSP recirculation line and ECCS minimum flow return lines to the RWSP, as well as the suction lines to each of the pumps. The NRC staff did not identify any scenarios where the probability of the RWSP failing had been substantially increased or steam binding of the pumps could occur.

Longer-term recovery actions were also considered. The initiation of the HPSI or LPSI pump provided additional time for isolating the shutdown cooling Train B suction line. Following reestablishment of RCS inventory, long-term cooling through Shutdown Cooling Train A or natural circulation were success paths.

These findings were considered qualitatively, along with previous shutdown cooling events, to assess the risk significance of this event. In particular, the loss of RCS inventory at the Wolf Creek Nuclear Station in 1994 was considered. This previous event resulted in the potential common-cause failure of the ECCS because the reactor coolant was diverted to the common suction of each of the safety injection pumps. This could have resulted in the steam binding of each of these pumps and significantly complicated recovery actions, had the leak not been promptly isolated. This common-cause failure potential, which contributed substantially to the significant risk at the Wolf Creek Nuclear Station, did not exist during the event at Waterford 3. However, recovery from the Waterford 3 event was largely driven by operator actions. These actions at the onset of the event were immediate and required very little diagnosis. Therefore, their success was given a high probability.

An event tree was developed with each of the significant recovery actions. These included isolation of LPSI Train B, initiation of HPSI Trains A or B or LPSI Train A within 10 minutes, isolation of LPSI Train B before heat up of the RWSP (several hours given makeup has been initiated), and establishment of natural circulation or shutdown cooling for long-term success. Additional recovery time could be made available with the recovery of the SITs. Based on minimal cutsets developed from the event tree and comparison to other loss of RCS inventory while-shutdown events, the NRC staff determined qualitatively that this event was of moderate risk significance. This was based on the failure probabilities for isolating the shutdown cooling suction, initiating makeup to the RCS, and establishing long-term cooling, which was dominated by operator action. The licensee performed a limited assessment of the event and verbally indicated that the event was of low risk significance. The licensee identified that additional risk assessment of this event would not be performed. The NRC staff will continue to assess the significance of this event in a quantitative manner.

c. Conclusions

The NRC staff concluded, from a qualitative assessment, that the event was of moderate risk significance. This finding was based on the required operator actions to isolate the shutdown cooling bypass to the refueling water storage pool and the initiation of RCS inventory makeup. The risk was minimized through the redundancy available to isolate the bypass leakage and the availability of both HPSI trains and LPSI Pump A.

E1.2 Licensee's Root Cause Analysis Investigation

a. Inspection Scope (93702)

The inspectors monitored the licensee's efforts to determine the root cause of the RCS drain-down event, including the affect on plant equipment and actions recommended to return equipment to an operable condition.

b. Observations and Findings

Following the RCS drain-down event, the licensee established a Significant Event Response Team (SERT) to evaluate the effect on plant equipment and recommend actions to correct any discrepancies. In addition, the licensee assigned a second group of operations and engineering personnel to identify the root cause of the event and recommend corrective actions. At the end of this inspection period, both the SERT and the root cause team had generated draft reports, which were essentially complete and included all the elements and conclusions required for the inspectors to assess the adequacy of these efforts.

The inspectors reviewed these reports and considered them to be adequate. Both included all relevant information presented in a logical manner and sufficiently supported the conclusions. The licensee's characterization of the event was based on available plant data, including plant monitoring computer data and information supplied by involved operations personnel. The inspectors requested and received the raw data used as a basis for the conclusions and recommendations to independently verify the adequacy of the licensee's efforts. No concerns were identified.

The root cause was identified as a roll pin that had come out of the reach-rod linkage, which caused the remote operator handwheel to become disconnected from the valve. This connection consisted of an inner cylindrical piece that fits into the center of an outer cylindrical piece. A hole was drilled through both pieces and the pin fitted into the hole to prevent movement relative to each other. The licensee measured the hole for the pin and determined that the inner piece diameter was slightly smaller than the outer piece diameter. With the pin installed, a reasonably tight fit would have been obtained with the inner piece but a clearance existed at each end with the outer piece. This difference caused impact forces to be exerted on the pin with each use of the remote handwheel. These impact forces caused the pin to move laterally over time with frequent use.

To correct the condition in the short term, the licensee installed a new pin in the linkage. The new pin would be subject to the same forces that caused the old pin to come out, but many cycles of the valve would be required for this occurrence. For the long term, the root cause investigation team recommended that either the hole be re-drilled to a uniform diameter or that the components be replaced to ensure that the impact forces would be eliminated. The inspectors considered these actions and recommendations adequate to ensure that the reach rod would perform satisfactorily in both the short and long terms.

Similar hole diameter conditions were found on several other holes on the same reach-rod operator. However, the associated pins were found to be intact and in their proper positions. All pins on this operator were replaced with new pins.

c. Conclusions

The licensee's efforts to assess the effect of the RCS drain-down event on plant equipment and to establish appropriate conclusions and recommendations was adequate. The conclusions and recommendations were based on appropriate data obtained from various sources. The root cause of the event was identified as a roll pin falling out of a reach-rod operator linkage due to the impact forces that it had been subjected to over many cycles of the valve. The licensee's actions and recommendations for short- and long-term corrective actions were adequate.

E2 Engineering Support of Facilities and Equipment

E2.1 Licensee's Assessment of the Condition of LPSI Train B

a. Inspection Scope (93702)

The inspectors reviewed the licensee's efforts to assess the condition of LPSI Train B, which was affected by the transfer of reactor coolant from the RCS to the RWSP.

b. Observations and Findings

Following the RCS drain down-event, the licensee performed a detailed evaluation and walkdown of LPSI Train B to assess the post-event condition and the readiness for declaring the system operable. The walkdown included a visual inspection of the piping, supports, and associated components for leaks or other potential damage resulting from water hammer, system misalignment, or transient event. Supports were inspected for broken U-bolts, anchor bolts, weld cracks, and signs of excessive deflection or damage. Other components were inspected for movement or damage. The walkdown identified no evidence of piping, pipe support, or component damage. The inspectors independently performed a walkdown of portions of the affected system and concluded that no evidence of damage was visible.

In addition, the licensee performed an evaluation of the temperature, pressure, and flow rate conditions in LPSI Train B during the transient. A review of the system design and the plant monitoring computer data indicated that the temperatures, pressures, and flow rates did not exceed design limits.

Prior to returning LPSI Train B to service, the licensee performed venting and ultrasonic testing of the system to ensure that no gases were trapped at the high points of the system. No gases or voids were identified during this procedure. In addition, LPSI Pump B was tested, using portions of the inservice testing procedure, to ensure adequate pump response. Also during this test, a vibrational survey was conducted, which indicated normal operation of the pump.

Following these walkdowns, inspections, evaluations, and tests, the licensee declared LPSI Train B operable. The inspectors considered these actions adequate to ensure proper operation of the system and had no concerns about the post-event condition of the individual components.

c. Conclusions

The licensee's actions to restore LPSI Train B to an operable status was adequate. Extensive walkdowns, inspections, evaluations, and testing were performed to ensure the integrity of the individual components and the system as a whole.

E2.2 Voiding in the RCS During the Drain-Down Event

a. Inspection Scope (93702)

The inspectors reviewed the licensee's evaluation and discussed, with selected engineering personnel, the potential for voiding in the RCS during the drain-down event.

b. Observations and Findings

The inspectors reviewed the licensee's evaluation of the potential for voiding to have occurred in the RCS during the drain-down event. This evaluation was included in the SERT root cause analysis report.

The licensee performed an as-found calibration check of the lowest reading wide-range pressure instrument, which indicated as low as 105 psia during the event. The instrument error was determined to be -0.4 psi. This indicated that the nominal pressure indicated on this instrument during the event was accurate. The licensee also considered instrument uncertainty for this instrument. This value was given as +37.8/-39.9 psia. Thus, pressurizer pressure potentially could have been as low as 68 psia considering applicable uncertainties. The highest reading representative core exit thermocouple indicated 308°F during the event. The saturation pressure for 308°F is approximately 75 psia. These numbers indicate that saturation conditions may have existed in the RCS during the event; therefore, voiding may have occurred. Based on the RCS temperature and pressure indication, the potential for saturated conditions existed for about 7 minutes.

During the event, the reactor vessel level monitoring system heated junction thermocouple indicated no void was formed in the upper reactor vessel head area. However, the response time of this instrument was relatively long, which would indicate that, if voiding did occur in the vessel head area, it was for a short duration time period. Licensee engineering personnel provided a figure of approximately 5700 gallons from the top of the vessel head to the top of the hot legs. This would be the volume of water required to be displaced in the head area to begin draining the steam generator tubes and inhibit establishment of single-phase, natural circulation conditions.

The licensee used two different methods to attempt to quantify the amount of possible void formation in the RCS. The first involved estimating the change in RWSP water volume as a result of the event. The RWSP level before the event was 85.25 percent. Due to dynamic effects in the RWSP during the event, the level indication fluctuated and finally stabilized at about 86.10 percent. Converting these levels to gallons and taking the difference yielded approximately 4788 gallons transferred into the RWSP during the event. To establish a bounding number, the licensee also used the maximum RWSP level indication experienced during the event of 86.49 percent. This level resulted in a maximum RWSP level increase of approximately 7000 gallons.

The known inventory in the RCS pressurizer and surge line prior to the event was determined to be approximately 4678 gallons. Subtracting this known volume from the RWSP level increase gives a range of 110 gallons or 15 cubic feet (using a 4788 gallon RWSP level increase) to 2322 gallons or 310 cubic feet (using a 7000 gallon RWSP level increase). This represents the potential size of the void formation in the RCS during the event.

The second method used the total charging system flow rate with two charging pumps operating, minus the letdown flow rate and the controlled bleed-off flow rate, to establish a net RCS injection flow rate. Using computer data, it was determined that a total of about 9 minutes passed between when the pressurizer level was at 5 percent cold cal and decreasing and the level returned to 5 percent cold cal as level was increasing. Multiplying the net RCS injection flow rate by 9 minutes gives 711 gallons injected into the RCS during these 9 minutes from a known starting point to a known ending point. Subtracting the known volumes in the pressurizer (from the 5 percent level) and surge line, a difference of approximately 97 gallons or 13 cubic feet was obtained. This represented the potential void volume. This method required some assumptions to be made, but was in reasonably good agreement with the first method’s results of 110 gallons.

The inspectors considered these methods as reasonable, given the information and data available to the engineering personnel. The resulting numbers were regarded as a gross measure of the potential voiding in the RCS rather than a precise calculation. Also, the time frame of potential voiding was relatively short since RCS pressure was increased quickly after Valve SI-407B was closed to stop the inventory loss. What voiding may have occurred did not hinder the establishment of single-phase, natural circulation or core cooling.

c. Conclusions

The licensee's methods for quantifying potential voiding in the RCS were reasonable. The resulting numbers indicated that a most probable value of approximately 97 to 110 gallons (13 to 15 cubic feet) of voiding may have occurred. The volume of voiding required to inhibit establishing single-phase, natural circulation conditions is approximately 5700 gallons.

IV. Plant Support

P1 Conduct of EP Activities

P3.1 Emergency Plan Implementation

a. Inspection Scope (93702)

The inspectors reviewed the licensee's actions in implementation of the emergency plan in response to the RCS drain-down event.

b. Observations and Findings

In response to the loss of pressurizer level and RCS pressure, the control room operators entered Procedure OP-901-111, then Procedure OP-901-131. The second step of Procedure OP-901-131 directed the shift supervisor to refer to Procedure EP-001-001, "Recognition and Classification of Emergency Conditions." At that time, the draindown had been stopped and pressurizer level and RCS pressure were slowly being recovered by the charging pumps.

The shift superintendent entered the classification procedure and noted that conditions had existed for approximately 1 minute that met the classification criteria for a Site Area Emergency, B/SAE/I: "Reactor Coolant System Leakage Greater than Charging Pump Capacity." Leakage from the RCS was greater than the capacity of all available charging pumps while Valve SI-407B was open, as indicated by rapidly lowering pressurizer level and RCS pressure. However, since the leakage to the RWSP had been stopped and there was no indication of additional RCS leakage, the shift superintendent decided declaration of a Site Area Emergency was inappropriate. The inspectors agreed that the Site Area Emergency declaration would have been inappropriate based on guidance provided in NUREG 1022, "Event Reporting Guidelines, 10 CFR 50.72 and 50.73," Revision 1, and EPPOS 2, "Emergency Preparedness Position 2." The licensee identified that the facility Emergency Plan did not address rapidly concluding events that exceeded an emergency action level, which presented a challenge to the shift superintendent to properly classify the event. The inspectors considered the lack of procedural guidance for short duration events as a challenge to consistent performance of operators in the future. The Emergency Preparedness Manager stated that guidance would be incorporated into the emergency plan implementing procedures for rapidly concluding events.

As the RCS pressure and pressurizer level were being recovered with HPSI flow, the shift superintendent observed that additional facility operators and management would be required on site to assist in full plant recovery, continuation to Mode 5, and evaluation of the status of systems affected by the draindown. At 5:33 a.m. on November 27, approximately 45 minutes after the start of the draindown, the shift supervisor assumed the duties of the Emergency Coordinator and declared an Alert based on EAL E/A/I: “Plant conditions exist that warrant precautionary activation of the TSC and placing the EOF and other key personnel on standby.” The inspectors agreed with the decision to declare an Alert as the appropriate classification.

State and local notifications were made in accordance with the appropriate procedures, although the Louisiana Office of Emergency Preparedness could not be contacted for 35 minutes after the Alert declaration due to communication problems at the state facility.

c. Conclusions

The control room staff properly implemented the facility Emergency Plan in response to the RCS drain down. The Alert declaration was appropriate and the required notification to state and local officials, as well as the NRC, were appropriate and timely.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on December 9, 1999, and on February 10, 2000. The licensee acknowledged the findings presented.

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

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. F. Burski, Director, Site Support
J. R. Douet, Manager, Plant Maintenance
C. M. Dugger, Vice-President, Operations
E. C. Ewing, Director, Nuclear Safety & Regulatory Affairs
R. M. Fili, Manager, Quality Assurance
C. Fugate, Operations Superintendent
J. G. Hoffpauir, Manager, Operations
J. D. Hunsaker, Manager, Site Support
T. R. Leonard, General Manager, Plant Operations
T. P. Lett, Superintendent, Radiation Protection
J. O'Hern, Manager, Training and Emergency Planning
E. Perkins, Jr., Manager, Licensing
L. N. Rushing, Manager, Mechanical and Civil Engineering
B. Thigpen, Director, Planning and Scheduling
A. J. Wrape, Director, Design Engineering

INSPECTION PROCEDURES USED

93702	Prompt Onsite Response to Events

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-382/9925-01	APP VIO	Failure to maintain LPSI Train B in operable condition (Section O1.2).
50-382/9925-02	APP VIO	Failure to follow procedure for position of Valve SI-147B

Closed

None

Discussed

None

LIST OF ACRONYMS USED

APP VIO	apparent violation
cal	calculation
CFR	Code of Federal Regulations
ECCS	emergency core cooling system
EOF	emergency offsite facility
gpm	gallons per minute
HPSI	high-pressure safety injection
LPSI	low-pressure safety injection
NRC	U.S. Nuclear Regulatory Commission
PDR	Public Document Room
PM	preventive maintenance
psi	pounds per square inch
psia	pounds per square inch absolute
RCS	reactor coolant system
RWSP	refueling water storage pool
SERT	Significant Event Response Team
SIT	safety injection tank
TS	Technical Specification
TSC	technical support center

LIST OF DOCUMENTS REVIEWED

Procedures:

OP-009-008, “Operating Procedure, Safety Injection System,” Revision 15
OP-010-004, “Power Operations,” Revision 0
OP-100-009, “Control of Valves and Breakers,” Revision 15
OP-100-001, “Duties and Responsibilities of Operators on Duty,” Revision 16
OP-901-111, “Reactor Coolant System Leak,” Revision 1
OP-901-131, “Shutdown Cooling Malfunction,” Revision 1
OP-902-000, “Emergency Entry Procedure,” Revision 8
OP-902-001, “Reactor Trip Recovery,” Revision 8
OP-902-002, “Loss of Coolant Accident Recovery,” Revision 8
OP-902-009, “Standard Appendices,” Revision 0
OP-903-026, “Emergency Core Cooling System Valve Lineup Verification,” Revision 9

UNT-004-044, “Component and Equipment Labeling,” Revision 0
UNT-005-010, “Independent Verification Program,” Revision 4
UNT-006-021, “Pump and Valve Inservice Testing,” Revision 3

Waterford 3 Steam Electric Station Emergency Plan, Revision 24
Waterford 3 Emergency Action Level Basis Document, Revision 3

Emergency Procedures:

EP-001-001 "Recognition and Classification of Emergency Conditions," Revision 18
EP-001-010 "Unusual Event," Revision 22
EP-001-020 "Alert," Revision 25
EP-001-030 "Site Area Emergency," Revision 24