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

December 19, 2003

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

SUBJECT: WATERFORD 3 STEAM ELECTRIC STATION - NRC TRIENNIAL FIRE PROTECTION INSPECTION REPORT 05000382/2003-011

Dear Mr. Venable:

On August 29, 2003, the Nuclear Regulatory Commission (NRC) completed the onsite portion of the subject inspection at your Waterford 3 Steam Electric Station. We discussed the preliminary results of the onsite inspection with you and members of your staff on August 29, 2003. Additional in-office inspection was performed during the weeks of September 1, September 15, September 29, October 13 , October 20, November 3, November 17, December 1, December 8, and December 15, 2003. You provided additional information regarding the findings to the team on September 15 and December 15, 2003. We held conference calls with your staff to discuss additional information on October 1 and 23, November 4, 20, and 21, and December 3, 9, 15, and 18, 2003. On November 21, 2003, we conducted a telephonic exit meeting with you and members of your staff to inform you of the results of the inspection. On December 3, 2003, by telephone we discussed with Mr. Gautam Sen and members of your staff a re-characterization of one of the findings from the November 21, 2003, exit. On December 19, 2003, we conducted a telephonic re-exit meeting with you and members of your staff, to inform you of the results of the inspection. The enclosed report presents the results of this inspection.

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

Based on the results of this inspection, the NRC has identified three violations that were evaluated under the risk significance determination process as having very low safety significance (Green). These findings have no immediate safety impact, as your staff took compensatory measures, which will remain in place until long-term corrective measures are implemented. These violations are being treated as non-cited, consistent with Section VI. A of the Enforcement Policy. The violations are described in the subject inspection report. If you contest the non-cited violations or significance of the non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to

the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Waterford 3 Steam Electric Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/ by RLN

Charles S. Marschall, Chief
Engineering and Maintenance Branch
Division of Reactor Safety

Docket: 50-382
License: NPF-38

Enclosure:
NRC Inspection Report
05000382/2003-011

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-382
License:	NPF-38
Report No.:	05000382/2003-011
Licensee:	Entergy Operations, Inc.
Facility:	Waterford 3 Steam Electric Station
Location:	17265 River Road Killona, Louisiana 70066-0751
Dates:	August 11 through December 19, 2003
Team Leader	J. L. Taylor, Senior Reactor Inspector Engineering and Maintenance Branch
Inspectors:	R. P. Mullikin, Senior Reactor Inspector Engineering and Maintenance Branch
	J. M. Mateychick, Reactor Inspector Engineering and Maintenance Branch
	G. B. Miller, Reactor Inspector Engineering and Maintenance Branch
Accompanying Personnel:	R. Deems, Brookhaven National Laboratory
Approved By:	Rebecca Nease, Team Leader Engineering and Maintenance Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR05000382/2003-011; 08/11-12/19/2003; Waterford 3 Steam Electric Station; Triennial Fire Protection Inspection.

The inspection was conducted by a team of four regional inspectors, and one accompanying contractor employee. Three Green non-cited violations were identified. The significance of issues is indicated by their color (Green, White, Yellow, Red) using NRC Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a violation of Technical Specification 6.8.1.f for failure to establish a procedure that implements the fire protection program. Specifically, the licensee failed to implement a procedure to functionally test certain electrical circuits on the emergency diesel generator mini-sequencer, which is relied upon for achieving shutdown in the event of a fire requiring control room evacuation and remote shutdown. Upon failure of this portion of the sequencer, automatic sequencing of certain components required for safe shutdown would be lost.

  This finding is greater than minor because it affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events (such as fire events) to prevent undesirable consequences (i.e., core damage). Specifically, automatic sequencing of certain safe shutdown equipment loads would be lost if this untested circuit failed. Additional operator actions to manually load these components could adversely affect the licensee's ability to perform other required safe shutdown manual actions within analyzed time requirements. The finding was determined to be of very low safety significance (Green) because the listed defense-in-depth elements in Figure 4-1 of Inspection Manual Chapter 0609, Appendix F, "Determining Potential Risk Significance Of Fire Protection and Post-Fire Safe Shutdown Inspection Findings," were not affected. In addition, (1) all other contacts on the switch are tested; (2) this particular mechanical switch has a very low failure probability; (3) control circuits of this type have a very low failure probability; and (4) operators have alternate verification steps in Procedure OP-901-502, "Evacuation of Control Room and Subsequent Plant Shutdown," to use in the event the mini-sequencer fails to sequence all the loads on the Emergency Diesel Generator B. (Section 1R05.4)

- Green. The team identified a violation of License Condition 2.C.(9) of the Waterford 3 Steam Electric Station Facility Operating License for failure to meet the fire protection program provision to provide adequate emergency lighting in areas needed for operation of safe shutdown equipment.

This finding is greater than minor because it affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events (such as fire events) to prevent undesirable consequences (i.e., core damage). Specifically, this finding adversely impacted the ability of the operators to align equipment necessary for safe shutdown of the plant in the event of a fire requiring evacuation of the control room. The finding is of very low safety significance (Green) because the listed defense-in-depth elements in Figure 4-1 of Inspection Manual Chapter 0609, Appendix F, "Determining Potential Risk Significance Of Fire Protection and Post-Fire Safe Shutdown Inspection Findings," were not affected. In addition, operators routinely carry flashlights, and additional flashlights are located in inventoried lockers at several plant locations. (Section 1R05.7)

- Green. The team identified two examples of a non-cited violation of License Condition 2.C.(9) of the Waterford 3 Facility Operating License for failure to identify and correct deficiencies adverse to fire protection, as required in the NRC-approved fire protection program.

  Example 1: The licensee failed to complete actions to correct conditions adverse to fire protection, in that, they inappropriately canceled a full-field verification test of their emergency lighting system. In response to NRC Generic Letter 86-10 and potential emergency lighting deficiencies documented in Condition Report CR-WF3-2000-0665, dated June 19, 2000, the licensee proposed a full-field verification test of the entire emergency lighting system to identify and correct any deficiencies. The team reviewed the licensee's basis for canceling the full-field verification test, and found it to be unacceptable.

  This finding is of greater than minor significance because it affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events (e.g., fire events) to prevent undesirable consequences (i.e., core damage). Specifically, this finding adversely impacted the operators' ability to align equipment necessary for safe shutdown of the plant in the event of a fire requiring the evacuation of the control room. The finding was determined to be of very low safety significance (Green), because the listed defense-in-depth elements in Figure 4-1 of Inspection Manual Chapter 0609, Appendix F, "Determining Potential Risk Significance Of Fire Protection and Post-Fire Safe Shutdown Inspection Findings," were not affected. (Section 4OA2)

  Example 2: The licensee failed to correct a deficiency in their methodology for determining if the emergency lighting system met the 10 CFR 50.65, Section (a)(1), maintenance rule goals. This led to the licensee's failure to identify that their emergency lighting system did not meet maintenance rule reliability criteria. Specifically, the licensee identified emergency lighting battery failures in condition reports; however, did not propose corrective actions to bring the emergency lighting system into compliance with maintenance rule reliability criteria.

This finding is of greater than minor significance since it was similar to Example 1.e of NRC Manual Chapter 0612, Appendix E, in that, portions of the emergency lighting system could not perform its intended safety function and its performance did not meet the established 10 CFR 50.65, Section a(1), goals. Specifically, this finding adversely impacted the operators' ability to align equipment necessary for safe shutdown of the plant in the event of a fire requiring the evacuation of the control room. The finding was determined to be of very low safety significance (Green) because the listed defense-in-depth elements in Figure 4-1 of Inspection Manual Chapter 0609, Appendix F, "Determining Potential Risk Significance Of Fire Protection and Post-Fire Safe Shutdown Inspection Findings," were not affected. (Section 4OA2)

REPORT DETAILS

1. REACTOR SAFETY

1R05 Fire Protection

The purpose of this inspection was to review the Waterford 3 Steam Electric Station fire protection program for selected risk-significant fire areas. Emphasis was placed on verification of the licensee's post-fire safe shutdown capability. The inspection was performed in accordance with the Nuclear Regulatory Commission (NRC) reactor oversight process using a risk-informed approach for selecting the fire areas and attributes to be inspected in accordance with Procedure 7111105T. Of the specified 3-5 area sample size, 4 were selected. The inspection scope was suspended for fire induced circuit failures of associated circuits while criteria for such review is the subject of an industry initiative. The team used the "Waterford 3 Station Individual Plant Examination of External Events," dated July 1995, to choose four risk-significant areas for detailed inspection and review. The fire areas chosen for review during this inspection were:

- RAB-2, heating and ventilation mechanical room
- RAB-7, relay room envelope
- RAB-8, switchgear room
- RAB-39, general plant areas (-35' and -4' levels)

For each of the selected fire areas, the team focused the inspection on the fire protection features and on the systems and equipment necessary for the licensee to achieve and maintain safe shutdown conditions in the event of a fire in those fire areas.

.1 Systems Required to Achieve and Maintain Post-Fire Safe Shutdown

a. Inspection Scope

The team reviewed the functional requirements identified by the licensee as necessary for achieving and maintaining hot shutdown conditions to ensure that at least one post-fire safe shutdown path was available in the event of a fire in each of the selected areas. The team reviewed piping and instrumentation diagrams of systems credited in accomplishing safe shutdown functions to independently verify whether the licensee’s shutdown methodology had properly identified the required components. The team focused on the following functions that must be ensured to achieve and maintain post-fire safe shutdown conditions.

- Reactivity control capable of achieving and maintaining cold shutdown reactivity conditions,

- Reactor coolant makeup capable of maintaining the reactor coolant inventory,

- Reactor heat removal capable of achieving and maintaining decay heat removal,

- Supporting systems capable of providing all other services necessary to permit extended operation of equipment necessary to achieve and maintain hot shutdown conditions.

A review was also conducted to ensure that all required electrical components in the selected systems were included in the licensee's safe shutdown analysis. The team identified the systems required for each of the primary safety functions necessary to shut down the reactor. These systems were then evaluated to identify the systems that interfaced with the fire areas inspected and were the most risk significant for reaching both hot and cold shutdown. The following systems were selected for review.

- Auxiliary feedwater system
- Chemical and volume control system system
- Reactor coolant system
- Reactor coolant gas vent system
- Safety injection system

Documents reviewed by the team are listed in Attachment 1.

b. Findings

No findings of significance were identified.

.2 Fire Protection of Safe Shutdown Capability

a. Inspection Scope

The team reviewed licensee documentation to verify that at least one post-fire safe shutdown success path was free of fire damage in the event of a fire in the selected fire areas. Specifically, the team examined the separation of safe shutdown cables, equipment, and components within the same fire areas. In addition, the team reviewed the protection of diagnostic instrumentation required for safe shutdown for fires in the selected areas. A sample of fire barrier penetration seals in the selected fire areas was reviewed by the team. The team reviewed the licensee's methodology for meeting the requirements of 10 CFR 50.48, and the bases for the NRC's acceptance of this methodology as documented in NRC safety evaluation reports. The team also reviewed license documentation, such as, the Waterford Updated Final Safety Evaluation Report, submittals made to the NRC by the licensee in support of the NRC's review of their fire protection program, and deviations from NRC regulations to verify that the licensee met license commitments.

b. Findings

No findings of significance were identified.

.3 Post-fire Safe Shutdown Circuit Analysis

a. Inspection Scope

On a sample basis, the team verified that cables of equipment required to achieve and maintain hot shutdown conditions in the event of fire in selected fire areas had been properly identified and either adequately protected from the potentially adverse effects of fire damage or analyzed to show that fire-induced faults (e.g., hot shorts, open circuits, and shorts to ground) would not prevent safe shutdown. During the inspection, a sample of redundant components associated with systems required to achieve and maintain hot shutdown conditions were selected for review. The sample included components associated with the auxiliary feedwater system, component cooling water system, and the essential service water system. From this list of components, the team reviewed cable routing data depicting the routing of power and control cables associated with each of the selected components. The team reviewed, on a sample basis, the analysis of electrical protective devices (e.g., circuit breaker, fuse, relay), coordination, and adequacy of electrical protection provided for nonessential cables, which share a common enclosure (e.g., cable trays) with cables of equipment required to achieve and maintain safe shutdown conditions.

b. Findings

No findings of significance were identified.

.4 Alternative Safe Shutdown Capability and Implementation

a. Inspection Scope

The team reviewed the licensee's alternative shutdown methodology to determine if the licensee has properly identified the components and systems necessary to achieve and maintain safe shutdown conditions from the remote shutdown panel and alternative shutdown locations for a fire in the unit’s control room. The team focused on the adequacy of the systems selected for reactivity control, reactor coolant makeup, reactor heat removal, process monitoring and support system functions. The team reviewed these systems to determine that hot and cold shutdown from outside the control room can be achieved and maintained with and without offsite power available.

The team reviewed documents to verify that the transfer of control from the control room to the alternative locations has been demonstrated to not be affected by fire-induced circuit faults by reviewing the alternative shutdown control circuits or the local controls available. Specifically, the team reviewed Functional Test Package MAI 401809 performed in 1999, in which, the licensee performed functional testing of Panel LCP-43 (remote shutdown panel) in accordance with Procedure OP-903-126, “Functional Testing of LCP-43,” Revision 1, Change 3. The purpose of the review was to verify that (1) the licensee conducts periodic operational tests of the alternative shutdown transfer capability and instrumentation and control functions and (2) the tests are adequate to show that if called upon the alternative shutdown capability would be functional upon transfer.

b. Findings

Introduction. The team identified a non-cited violation of very low safety significance (Green) for failure to test certain contacts of circuits affecting safe shutdown equipment. This is a violation of Technical Specification 6.8.1.f for failure to establish a procedure that implements the fire protection program. Specifically, the licensee failed to implement a procedure to test a portion of the circuitry of the emergency diesel generator mini sequencer, which is used in a control room evacuation event.

Description. The team identified that not all circuits of the Emergency Diesel Generator B mini-sequencer undergo periodic functional testing. The mini-sequencer loads four critical components upon the start of Emergency Diesel Generator B. The relays in the circuits are individually tested every 36 months; however, certain Fire Switch FR-2 contacts in the mini-sequencer circuits are not tested and the control circuits are not functionally tested. Failure of one of these circuits would result in an open control circuit causing the mini sequencer to fail to load (1) Component Cooling Water Pump B; (2) Diesel Generator B room Exhaust Fan E28; (3) Station Service Water Transformer 3B32; and (4) Station Service Transformer 3B315 on the protected emergency diesel generator. The loss-of-automatic sequencing of these loads on Emergency Diesel Generator B would require operators to load each of these systems manually on the diesel, possibly affecting their ability to perform other manual actions within the safe shutdown time-line.

Analysis. The team determined that the finding was of greater than minor significance because it affects the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of equipment that responds to initiating events to prevent undesirable consequences (i.e., core damage). The senior reactor analyst and the team leader evaluated this finding using the Phase 1 Significance Determination Process, as described in Inspection Manual Chapter 0609, Appendix F, "Determining Potential Risk Significance Of Fire Protection and Post-Fire Safe Shutdown Inspection Findings." Using Figure 4-1, Step 1 of Phase 1 in Appendix F, this finding was determined to be of very low safety significance (Green). The finding screened out as Green because the listed defense-in-depth elements in Figure 4-1 were not affected. In addition, (1) all other contacts on the switch are tested; (2) this particular mechanical switch has a very low failure probability; (3) control circuits of this type have a very low failure probability; and (4) operators have alternate verification steps in Procedure OP-901-502, "Evacuation of Control Room and Subsequent Plant Shutdown," to use in the event the mini-sequencer fails to sequence all the loads on the Emergency Diesel Generator B.

Enforcement. Technical Specification 6.8.1.f requires that the licensee establish procedures, which implement their fire protection program. The licensee's fire protection program, defined in License Condition 2.C.(9) of the Waterford 3 Facility Operating License, includes the Waterford 3 Final Safety Analysis Report. Section 9.5.1.3.1.C.5 of the Waterford 3 Final Safety Analysis Report states that the criteria for tests and test control in Appendix A to APCSB Branch Technical Position 9.5-1, have been considered and are documented in the Quality Assurance Program Manual and in the fire protection program.

APCSB Branch Technical Position 9.5-1 states, in part, that a test program should be established and implemented to assure that testing is performed to demonstrate conformance with design and system readiness requirements. The tests should be performed in accordance with written test procedures. Contrary to this statement, the licensee failed to implement a procedure to test certain contacts required to function in the event a fire requires control room evacuation and remote shutdown. Because the failure to test the circuit is of very low safety significance and has been entered into the corrective action program as Condition Report CR-WF3-2003-02441, this violation is being treated as a non-cited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000382/2003011-01, Failure to Test Certain Emergency Diesel Generator B Mini-Sequencer Contacts.

.5 Operational Implementation of Alternative Shutdown Capacity

a. Inspection Scope

The team reviewed operator training lesson plans, classroom materials, and training attendance records for licensed and non-licensed personnel to determine if it included the alternative safe shutdown capability.

The team reviewed licensee staffing to determine if personnel required to achieve and maintain the plant in hot shutdown following a fire using the alternative shutdown system can be provided from normal onsite staff, exclusive of the fire brigade. The team verified this by reviewing Procedure OP-901-502, "Evacuation of Control Room and Subsequent Plant Shutdown," and the licensee's commitments for minimum shift staffing contained in Procedure OP-100-001, "Operations Standards and Management Expectations."

The team performed walkdowns with licensed and non-licensed operators of the actions defined in Procedure OP-901-502. Procedure OP-901-502 is used by operators for performing an alternative shutdown from the remote shutdown panel and from local control stations in the plant. The team observed a timed field walkthrough simulating the performance of Procedure OP-901-502 by the licensee operations personnel. The team also reviewed equipment labeling to determine if it was consistent with the procedure.

b. Findings

Introduction. The team identified an unresolved item involving the adequacy of Procedure OP-901-502, "Evacuation of Control Room and Subsequent Plant Shutdown," which is used by operators to perform actions necessary for achieving and maintaining safe shutdown conditions within established times. This item is unresolved pending resolution of the differences between team observations of a field walkthrough of OP-901-502 and the licensee's basis for concluding that operators could perform the actions within established times.

Description. The team reviewed Procedure OP-901-502, "Evacuation of Control Room and Subsequent Plant Shutdown," which implements the alternative safe shutdown capability for fires in the control room or cable spreading room requiring control room

evacuation and remote shutdown. The team also reviewed Calculation EC-F00-026, "Post Fire Safe Shutdown Analysis," which defines (1) the systems and functions necessary for achieving alternative shutdown; (2) the actions required to operate the systems to establish those functions; and (3) the times in which these functions must be established in order to safely achieve and maintain hot shutdown conditions, as defined in 10 CFR Part 50, Appendix R, Section III.L. Section III.L establishes performance goals for achieving and maintaining hot shutdown conditions using alternative shutdown methodology.

The licensee performed a timed field walkthrough of Procedure OP-901-502, which was observed by the team. The team observed that operators did not perform the actions in the procedure within the times analyzed in Calculation EC-F00-026. Specifically, the team noted the following discrepancies between the timed walkthrough of Procedure OP-901-502 and the times established by the fire safe shutdown analysis (Calculation EC-F00-026):

OPERATOR ACTION	TIMES ESTABLISHED BY ANALYSIS	FIELD WALKTHROUGH TIMES
Isolate MSIVs	10 min.	13 min.
Secure RCPs	10 min.	22 min.
Establish Train B Electric Power	20 min.	37 min.
Establish EFW	25 min.	44 min.
Isolate Main Steam Line Drains from SGs	30 min.	41 min.
De-Energize Train A Electrical Power	30 min.	58 min.
RCS Inventory Control & Makeup via Charging Pump B	60 min.	65 min.
Place fire isolation switches in isolate	10 min.	13 min.
Place LCP-80 key switches in isolate	10 min.	14 min.
Open breakers on PDP 91B	25 min.	26 min.
Verify closed breaker to align CC valves	20 min.	28 min.
Establish B SWGR room ventilation	20 min.	37 min.
Disable EDG A	30 min.	58 min.
Restore Press Proportional Heater Bank 2	60 min.	67 min

The team found that the procedure did not identify the time-critical actions so that operators could make informed decisions concerning which actions to take first, based on plant conditions. Operators are expected to prioritize the order of performance of the

procedural steps based on the current plant conditions during the fire. In addition, operator training on alternative shutdown did not provide operators with information on the time-critical actions established in Calculation EC-F00-026. The team also found that in issuing Procedure OP-901-502 and subsequent revisions, the licensee had not formally verified that the procedure could be implemented in a manner that satisfied the time requirements established in the fire safe shutdown analysis (Calculation EC-F00-026).

As immediate corrective action, the licensee briefed each operating crew concerning this issue, and which actions were considered to be time-critical. The licensee also revised Procedure OP-901-502 to add notes identifying which steps were time-critical (performed in less than 1 hour), and describing the conditions under which the time limitation applied. In addition, the licensee added an attachment to Procedure OP-901-502 to provide operators with a quick reference for actions required to be performed within 30 minutes. This finding was entered into the licensee's corrective action program as Condition Report CR-WF-2003-02440. On December 13, 2003, the licensee provided documentation of a timed field walkthrough of the revised procedure in which operators were able to perform manual actions in accordance with the safe shutdown analysis (Calculation EC-F00-026). Therefore, no current safety concern exists with respect to this unresolved item.

The licensee maintained that the times recorded in the simulated field walkthrough were inflated due to the artificiality of the scenario and inappropriate cues to the operators. In support of this view, on September 15, 2003, the licensee provided a qualitative assessment that considered the effects the field walkthrough artificiality had on the measured timing of the operator actions. In this assessment the licensee concluded that in considering the artificially-inflated times, operators could perform the actions within the times established by the safe shutdown analysis. The team reviewed this assessment, and found differences between walkthrough information used in the assessment and observations made by the team during the walkthrough. For instance, during the walkthrough, the team noted that the timekeepers had, at times, stopped their stopwatches when excessive time was taken for actions that would not be required during an actual scenario, such as processing through the radiologically controlled area access. In addition, the team noted that the licensee did not re-perform or re-time any of the manual actions in the field to substantiate their conclusion with actual field data. The team leader concluded that to fully understand the basis of the licensee's conclusion, additional review and discussion is necessary to resolve the differences between the licensee's qualitative assessment and team observations. Therefore, this issue is unresolved pending further review and discussion of the licensee's basis for their conclusion in the assessment. The licensee's qualitative assessment is included as Attachment 2 to this report.

Analysis. The risk of this issue will be evaluated if, upon further review, the team determines that this issue is a violation.

Enforcement. Technical Specification 6.8.1.f requires, in part, that the licensee establish procedures that implement their NRC-approved fire protection program. Procedure OP-901-502 and Calculation EC-F00-026 implement the licensee's fire protection alternative shutdown methodology for a fire event requiring control room

evacuation and remote shutdown. During a walkthrough of Procedure OP-901-502, the team observed that certain operator actions were not be performed within times established by the safe shutdown analysis (Calculation EC-F00-026). As discussed above, the licensee submitted a qualitative assessment, in which they concluded that operators would be able to take actions within established times. Whether this issue is a violation remains unresolved pending resolution of the differences between the licensee's assessment and the team's observations: URI 050000382/2003011-02, Concerns Regarding the Timing of Operator Actions Performed Using the Implementing Procedure for Alternative Shutdown Capability.

.6 Emergency Communications

a. Inspection Scope

The team reviewed the communications required to implement Procedure OP-901-502, "Evacuation of Control Room and Subsequent Plant Shutdown." The plant maintenance radio system is used by operations to perform an alternative shutdown outside of the control room. Operations communicates via portable radios carried by the auxiliary operators and the operators at the remote shutdown panel. Sound powered phone connections provide communications between the +35 relay room and the remote shutdown panel area. The team verified that (1) portable radios and sound powered phone connections are available, operable, and adequate for the performance of safe shutdown functions; (2) the designs of the communication systems assure that damage from a control room fire will not impact the performance of the rest of the system; and (3) the radio and sound powered phone systems are sufficient to perform the required communications identified in the alternative shutdown procedure.

b. Findings

No findings of significance were identified.

.7 Emergency Lighting

a. Inspection Scope

The team reviewed the emergency lighting system required for safe shutdown activities to verify that it was adequate for supporting the performance of manual actions required to achieve and maintain hot shutdown conditions, and for illuminating access and egress routes to the areas where manual actions are required. The team reviewed test procedures, test data, and vendor documentation to verify the individual battery operated units were able to supply light for the required 8-hour period.

b. Findings

Inadequate Emergency Lighting for Supporting Operator Actions

Introduction. The team identified a non-cited violation of very low safety significance (Green) for failure to provide emergency lighting that complies with 10 CFR Part 50, Appendix R, Section III.J, a provision of the Waterford 3 Steam Electric Station, fire

protection program. This is a violation of License Condition 2.C.(9) of the Waterford 3 Steam Electric Station Facility Operating License, which requires that the licensee maintain all provisions of their fire protection program.

Description. During the week of August 11, 2003, the team conducted field walkdowns of the areas in which safe shutdown equipment was required to be operated, and access and egress routes thereto. The team identified 12 locations where the installed emergency lighting appeared to be inadequate to support operator actions in performance of Procedure OP-901-502, "Evacuation of Control Room and Subsequent Plant Shutdown." The licensee staff initiated Condition Report CR-WF3-2003-2286 to address the team's concerns. After thoroughly reviewing the lighting in these areas, the licensee concluded that emergency lighting in three of the identified areas would not be sufficient to allow an operator to perform the necessary safe shutdown functions in the event of a fire requiring the evacuation of the control room. The areas with inadequate emergency lighting and the potential consequences are described below.

- Turbine-driven Emergency Feedwater Pump on the -35 Elevation of the Reactor Auxiliary Building: The installed emergency lighting was partially obstructed by piping and too distant to provide sufficient lighting to perform manual alignment of the pump in accordance with Step 16 of Procedure OP-901-502, "Evacuation of Control Room and Subsequent Plant Shutdown." Operators may not be able to realign the pump in time to prevent boiling the steam generators dry if the pump trips during a fire event.

- Station Uninterruptible Power Supply, Switchgear Panel in Switchgear Room AB: No emergency lighting is provided for operators to strip loads from the switchgear panel to extend station battery availability in accordance with Step 12 of Procedure OP-901-502. The inability to strip loads from the station batteries may shorten the life of the batteries to less than the 4 hours established by the safe shutdown analysis. This panel is located in close proximity to two Appendix R lockers with inventory-controlled flashlights.

- Permissive to Close Keyswitch for the Feeder to Pressurizer Proportional Heater Bank 2 in Switchgear Room B: No emergency lighting is provided for operators to operate this switch as required in Procedure OP-901-502. In the event that 480-V Switchgear 32B is unavailable, the inability to close the pressurizer proportional heater Bank 2 feeder breaker could adversely impact primary plant pressure control. This key switch is located in close proximity to two Appendix R lockers with inventory-controlled flashlights.

As an immediate corrective action, the licensee staged additional boxes of flashlights with spare batteries in the plant and labeled them for emergency use only. In addition, operators were notified of the flashlight locations and purpose. To evaluate the extent of this finding, the licensee also performed a field test of the emergency lighting system in which they identified an additional ten areas with inadequate emergency lighting. The licensee entered all 13 examples of inadequate lighting into the corrective action program as Condition Report CR-WF3-2003-2735.

Analysis. This finding is greater than minor because it affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events (such as fire events) to prevent undesirable consequences (i.e., core damage). Specifically, this finding adversely impacted the ability of operators to align equipment necessary for safe shutdown of the plant in the event of a fire requiring evacuation of the control room. The senior reactor analyst and the team leader evaluated this finding using the Phase 1 Significance Determination Process as described in NRC Inspection Manual Chapter 0609, Appendix F, "Determining Potential Risk Significance Of Fire Protection and Post-Fire Safe Shutdown Inspection Findings." Using Figure 4-1, Step 1 of Phase 1 in Appendix F, this finding was determined to be of very low safety significance (Green) because the listed defense-in-depth elements in Figure 4-1 were not affected. In addition, operators routinely carrying flashlights and inventory-controlled additional flashlights are available in the Appendix R lockers.

Enforcement. License Condition 2.C.(9) of the Waterford 3 Steam Electric Station Facility Operating License states that the licensee shall implement and maintain in effect all provisions of licensee's fire protection program, part of which is described in Supplement 3 of NUREG-0787, "Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit 3," dated July 1981. Supplement 3 stated that the licensee committed to installing self-contained 8-hour battery pack emergency lighting in all areas of the plant which must be manned to bring the plant to a safe cold shutdown condition and in access and egress thereto. Supplement 3 further stated that with this commitment, the licensee satisfied the requirements of 10 CFR Part 50, Appendix R, Section III.J. Contrary to this statement, the licensee failed to provide adequate emergency lighting for all local actions necessary for bringing the plant to a safe shutdown, as described in Procedure OP-901-502, "Evacuation of Control Room and Subsequent Plant Shutdown." This is a violation of License Condition 2.C.(9) of the Waterford 3 Steam Electric Station Facility Operating License. Because the failure to provide adequate emergency lighting is of very low safety significance (Green) and the inadequacies have been entered into the corrective action program (CR-WF3-2003-2286 and CR-WF3-2003-2735), this violation is being treated as a non-cited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000382/2003011-03, Inadequate Emergency Lighting for Supporting Operator Actions.

.8 Cold Shutdown Repairs

a. Inspection Scope

The team reviewed the licensee’s safe shutdown analysis to determine whether repairs were required to achieve cold shutdown and to verify that the required material was available. The licensee’s safe shutdown analysis states that the only repairs necessary to achieve cold shutdown are the replacement of fuses.

b. Findings

No findings of significance were identified.

.9 Compensatory Measures

a. Inspection Scope

The team verified, by sampling, that adequate compensatory measures were put in place by the licensee for out-of-service, degraded, or inoperable fire protection and post-fire safe shutdown equipment, systems or features (e.g., detection and suppression systems, or passive fire barrier features). The team also verified that the fire system impairments were being corrected in a timely manner.

b. Findings

No findings of significance were identified.

.10 Fire Protection Systems, Features, and Equipment

a. For the selected fire areas, the team evaluated the adequacy of fire protection features, such as fire suppression and detection systems, fire area barriers, penetration seals, and fire doors. To do this, the team observed the material condition and configuration of the installed fire detection and suppression systems, fire barriers, and construction details and supporting fire tests for the installed fire barriers. In addition, the team reviewed license documentation, such as NRC safety evaluation reports, and deviations from NRC regulations and the National Fire Protection Association codes to verify that fire protection features met license commitments. Documents reviewed by the team are listed in Attachment 1.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

a. Inspection Scope

The team selected a sample of the licensee’s condition reports associated with the Waterford Steam Electric Station, Unit 3, fire protection program to verify that the licensee had an appropriate threshold for identifying deficiencies. The team also reviewed several licensee audits of the fire protection program to evaluate the licensee's ability to identify deficiencies. In addition, the team reviewed the corrective actions proposed and implemented to verify that they were effective in correcting the identified deficiencies.

b. Findings

Introduction. The team identified two examples of a non-cited violation of very low safety significance (Green) for failure to correct conditions adverse to fire protection, which is a provision of the Waterford fire protection program. This is a violation License Condition 2.C.9 of the Waterford 3 Steam Electric Station Facility Operating License.

Description.

Example 1 - Inadequate Corrective Actions for Identified Emergency Lighting System Inadequacies: The first example of inadequate corrective action involved the canceling of a full field verification test of emergency lighting, which was proposed in response to Generic Letter 86-10, “Implementation of Fire Protection Requirements.” On June 19, 2000, the licensee initiated Condition Report CR-WF3-2000-0665 to address the failure to field test the emergency lighting system to ensure that adequate emergency lighting was available to meet Section III.J of 10 CFR Part 50, Appendix R. As part of the corrective action, the licensee proposed developing a procedure and conducting a full field test of the installed emergency lighting to verify that it could perform its function to provide lighting in support of safe shutdown operator actions.

During an NRC triennial fire protection inspection performed in October 2000 (NRC Inspection Report 50-382/00-07, dated November 29, 2003), the team questioned the adequacy of the emergency lighting in some locations of the plant. The team was provided with Condition Report CR-WF3-2000-0665, in which the licensee had already identified potential areas where emergency lighting might be inadequate. As corrective action, the licensee had proposed a full field verification test of the entire emergency lighting system to identify and correct any deficiencies.

The special test instruction to perform the field verification was completed and approved for use on May 15, 2002. The test, which was originally scheduled to be performed on August 1, 2002, was canceled and rescheduled five times between August 2002 and May 2003. On July 3, 2003, in lieu of performing this testing, the licensee developed an evaluation to justify permanently canceling the full-field verification. In this evaluation, the licensee cited the lack of findings in NRC inspection reports (including NRC Triennial Fire Protection Inspection Report 50-382/00-07). The licensee concluded that the lack of emergency lighting findings in past NRC inspection reports implied that their installed emergency lighting system was adequate. Section 3.3 of NUREG 1409, "Backfitting Guidelines," states that simply not challenging a licensee's practice in an inspection report would not be considered approval. The team found the licensee's basis for canceling the corrective actions to field test their emergency lighting to be inadequate. The licensee initiated Condition Report CR-WF3-2003-2455 to address the cancellation of the full field verification test. In addition, the licensee initiated Condition Report CR-WF3-2003-2445 to readdress the concern described earlier in Condition Report CR-WF3-2000-0665, which was inappropriately closed.

Example 2 - Failure to Take Corrective Actions to Address an Inadequate Maintenance Rule Emergency Lighting Functional Failure Definition

The second example of inadequate corrective action involved the failure to correct a deficiency in the methodology for determining if the emergency lighting system met the 10 CFR 50.65, Section (a)(1), maintenance rule goals. This contributed to the licensee's inability to identify that multiple failures of the emergency lighting batteries resulted in not meeting maintenance rule reliability criteria. The licensee identified the emergency lighting battery failures in condition reports, but because they erroneously concluded that they met maintenance rule reliability criteria, proposed no corrective actions.

On September 5, 2000, the licensee initiated Condition Report CR-WF3-2000-1026 to address multiple failures of the 8-hour discharge test for the emergency lighting batteries installed at the facility. Corrective actions for Condition Report CR-WF3-2000-1026 included requiring condition reports to be written for every emergency light failure and developing a schedule for periodic battery and lamp replacement. As part of the corrective action, the licensee performed an evaluation to determine if these emergency lighting battery failures exceeded the maintenance rule reliability criteria established for emergency lighting. This evaluation, documented in the Root Cause Determination Report dated February 13, 2001, stated, "The functional failure definition is vague in that there is no specified time range. Systems Engineering has interpreted the definition to require 90% availability at any one time. Because only approximately 5 emergency battery lights (approximately 2%) are tested at any one time, it is not likely the system will ever experience a functional failure as currently defined and interpreted." The report further stated, "the Maintenance Rule functional failure definition is not adequate to evaluate the performance of the system." The team found that in Condition Report CR-WF3-2000-1026, the licensee identified that the maintenance rule functional failure definition was not adequate for evaluating emergency lighting system performance; however, no actions to correct this deficiency were taken.

The team reviewed the results from 8-hour emergency light discharge tests conducted from August 14, 2002, to October 16, 2002. The tests consisted of unplugging the emergency light unit and ensuring the lamps came on and remained lit for 8 hours. Of the 25 emergency lights tested during this 2 month period, 10 failed the discharge test, which the team determined to be a 40 percent failure rate, corresponding to a 60 percent acceptance rate. However, the maintenance rule reliability acceptance criteria was a rate of 90 percent. Although condition reports were written for each of the failures, the condition reports listed only the number of failures with no mention of the number of emergency lights tested, or the failure rate. The team found that in Condition Report CR-WF3-2000-1026, the licensee failed to correct an identified deficiency in the maintenance rule functional failure definition for evaluating emergency lighting system performance. This resulted in emergency lighting battery failures not being properly evaluated against maintenance rule acceptance criteria. Consequently, the licensee failed to recognize that the emergency lighting system did not meet established maintenance rule 10 CFR 50.65, Section (a)(1), goals.

The licensee staff initiated Condition Report CR-WF3-2003-2439 to address the team's concerns regarding the ability of the emergency lighting system to provide its function to support operator actions in the event normal lighting is unavailable. As interim corrective action, boxes of flashlights with spare batteries were labeled for emergency use only and staged in the plant, and operators were notified of their locations and purpose.

Analysis. The team determined that both examples of inadequate corrective actions were greater than minor. The first example is greater than minor because it affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events (such as fire events) to prevent undesirable consequences (i.e., core damage). Specifically, this finding adversely impacted the ability of operators to align equipment necessary for safe shutdown of the plant in the event of a fire requiring evacuation of the control room. This example of inadequate corrective action involves the inappropriate canceling of the emergency lighting system full-field verification test as proposed in a condition report. This resulted in the licensee's failure to identify deficiencies in their emergency lighting system.

The second example involves the failure to correct a deficiency in the maintenance rule emergency lighting functional failure definition, which led to the failure to identify that the emergency lighting system did not meet the maintenance rule reliability criteria. This second example was greater than minor since it was similar to Example 1.e of NRC Manual Chapter 0612, Appendix E, in that portions of the emergency lighting system could not perform its intended safety function and its performance did not meet the established 10 CFR 50.65, Section a(1), goals.

The senior reactor analyst and the team leader evaluated both examples of inadequate corrective actions using the Phase 1 Significance Determination Process as described in Inspection Manual Chapter 0609, Appendix F, "Determining Potential Risk Significance Of Fire Protection and Post-Fire Safe Shutdown Inspection Findings." Using Figure 4-1, Step 1 of Phase 1 in Appendix F, this finding was determined to be of very low safety significance (Green). The finding was screened out as Green because the listed defense-in-depth elements in Figure 4-1 were not affected. In addition, operators routinely carrying flashlights and inventory-controlled additional flashlights are available in the Appendix R lockers.

Enforcement. License Condition 2.C.9 of the Waterford 3 Steam Electric Station Facility Operating License requires the licensee to implement and maintain all provisions of their approved fire protection program as described, in part, in the Final Safety Analysis Report for the facility. Section 9.5.1.3.1.C.8 of the Waterford Steam Electric Station, Unit 3, Final Safety Analysis Report states that criteria for corrective action in Appendix A to APCSB Branch Technical Position 9.5-1 have been considered and are documented in the Quality Assurance Program Manual and fire protection program. Branch Technical Position 9.5-1, "Guidelines for Fire Protection at Nuclear Power Plants," states that measures should be established to assure that conditions adverse to fire protection are promptly identified, reported, and corrected. Contrary to this requirement, the licensee inappropriately canceled corrective actions (that called for a field verification of emergency lighting) to correct identified conditions adverse to fire

protection. This is one example of a violation of License Condition 2.C.9 of the Waterford 3 Steam Electric Station Facility Operating License.

In addition, the licensee failed to correct a deficiency in their methodology for determining if the emergency lighting system met the 10 CFR 50.65, Section (a)(1), maintenance rule goals. This led to the licensee's failure to recognize that emergency lighting battery failures resulted in their emergency lighting system not meeting maintenance rule reliability criteria. This is the second example of a violation of License Condition 2.C.9 of the Waterford 3 Steam Electric Station Facility Operating License.

In both examples, the licensee failed to correct identified deficiencies to assure that the emergency lighting system would be capable of performing its intended function. This is a violation of License Condition 2.C.9 of the Waterford Steam Electric Station Facility Operating License. Because both examples of this violation are of very low safety significance and have been entered into the corrective action program (CR-WF3-2003-2439, -2445, and -2455), this violation is being treated as a non-cited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000382/2003011-04, Inadequate Corrective Actions for Deficiencies Associated with the Emergency Lighting System, Two Examples.

4OA6 Meetings, including Exit

On August 29, 2003, at the conclusion of the team's onsite inspection, the team leader debriefed Mr. Joseph E. Venable, Vice President Operations, and other licensee management and staff members on the preliminary results of the inspection.

On September 10, 2003, the team leader conducted a telephone meeting with Mr. Gautam Sen and other licensee staff members, during which the issues for further clarifying information was to be provided, were discussed. The team leader agreed to an extension of time for submittal of the additional information until September 14, 2003. The additional information (Attachment 2) was received September 15, 2003, and review by staff ensued.

On October 1, 2003, the team leader and other members of the inspection team conducted an telephonic conference call with Mr. Gautam Sen and other members of the licensee's management and staff (at the licensee's request) to discuss the additional information forwarded to the team and to further clarify the inspection findings and the bases therefor. Additional discussions were held October 23, November 4, 20, and 21, 2003.

On November 21, 2003, the team leader conducted a telephonic exit meeting with Mr. Joseph E. Venable, Vice President Operations, and other licensee management and staff members to inform them of the results of the inspection.

On December 3, 2003, a conference call with Mr. Gautam Sen and other licensee management and staff members was held to discuss the possible re-characterization of one of the findings from November 21, 2003, exit. Additional conference calls were held

with licensee management and staff on December 9, 15, and 18, 2003, to clarify the NRC's findings and the licensee's corrective actions.

On December 19, 2003, the team leader conducted a telephonic re-exit meeting with Mr. Joseph E. Venable, Vice President Operations, and other licensee management and staff members to re-characterize one of the team's findings. The licensee was asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Brumfield, QA Manager
J. Burke, Quality Assurance Engineer
B. Collyer, Fire Protection Engineer
R. Douet, General Manager-Plant Operations
A. Holder, Fire Protection Engineer
C. Lambert, Director-Engineering
H. Lewis, Licensed Operator Requalification Lead
M. Melancon, Safety Analysis Engineer
W. Pendergrass, Operations
K. Peters, Director, NSA
O. Pipkins, Licensing Engineer
K. Renau, System Engineer
G. Sen, Licensing Manager
R. Simpson, Operations
T. Tankersley, Training Manager
J. Venable, Vice President Operations

NRC

M. Hay, Senior Resident Inspector
M. Runyan, Senior Reactor Analyst
T. Pruett, Senior Reactor Analyst
P. Qualls, Office of Nuclear Reactor Regulation

ITEMS OPENED AND CLOSED

Opened

05000382/03011-02	URI	Concerns Regarding the Timing of Operator Actions Performed Using the Implementing Procedure for Alternative Shutdown Capability. (Section 1R05.5)

Opened and Closed

05000382/03011-01	NCV	Failure to Test Certain Emergency Diesel Generator "B" Mini-Sequencer Contacts (Section 1R05.4)
05000382/03011-03	NCV	Inadequate Emergency Lighting for Supporting Operator Actions (Section 1R05.7)

05000382/03-11-04	NCV	Inadequate Corrective Actions for Deficiencies Associated with the Emergency Lighting System, Two Examples. (Section 4OA2)

DOCUMENTS REVIEWED

The teams selected and reviewed the following documents to accomplish the objectives and scope of the inspection and to support any findings:

AUDITS/ASSESSMENTS

Number	Description	Date
W3H3-2001-0045	Quality Assurance Audit of "Fire Protection"	April 4, 2001
W3H3-2002-0031	Quality Assurance Audit of "Fire Protection"	March 4, 2002
W3H3-2003-005	Quality Assurance Audit of "Fire Protection"	February 28, 2003

CALCULATIONS

Number	Title	Revision/ Date
EC-E91-056	Relay Coordination Study	1
EC-F00-026	Post Fire Safe Shutdown Analysis (Appendix R Revalidation Project)	1
ER-W3-2003-0161-100	Reroute Divisional Cable for the Chilled Water System in Fire Area RAB-2	0
ER-W3-2001-0122-000	Minimum Time for Restoring Essential Chilled Water Flow	November 15, 2001
MN(Q)-3-5	Flooding Analysis Outside Containment	3

CONDITION REPORTS

CR-WF3-2000-00665	CR-WF3-2000-01026
CR-WF3-2000-01126	CR-WF3-2001-00144
CR-WF3-2001-00157	CR-WF3-2002-01639
CR-WF3-2003-00002	CR-WF3-2003-02286
CR-WF3-2003-02439	CR-WF3-2003-02440
CR-WF3-2003-02441	CR-WF3-2003-02445
CR-WF3-2003-02446	CR-WF3-2003-02448
CR-WF3-2003-02451	CR-WF3-2003-02455

DRAWINGS

Number	Title	Revision
G-172	Flow Diagram - Reactor Coolant System	30
G-247 S06	Communication Riser Diagram - Sh 1	7
G-285	Main One Line Diagram	15
G-286	Key Auxiliary One Line Diagram	16
G-287, Sheet 1	125 VDC and 120 VAC One Line Diagram	19
G-287. Sheet 2	125 VDC and 120 VAC One Line Diagram	2
G-1357	Reactor Auxiliary Bldg. Plan El. -35.00'	1
G-1359	Reactor Auxiliary Bldg. Plan El. +21.00'	2
G-1360	Reactor Auxiliary Bldg. Plan El. +46.00'	1
G-1368	Reactor Bldg. & Wing Area Plan El. +21.00'	1
G-1369	Reactor Bldg. & Wing Area Plan El. +46.00'	0
G-1375	Reactor Auxiliary Bldg. Plan El. +35.00'	1
G-168, SH 1&2	Chemical and Volume Control System	48
G-151, SH 1	Main & Extraction Steam System	38
G-167, SH 1,2,3&4	Safety Injection System	44
G-583, SH 1,2,3&4	HVAC Chilled Water System	21
G-160, SH 1,2,3,4,5&6	Component Cooling Water System	49
G-853S15 thru S2379	HVAC Airflow Diagrams	5
G-FP-0017	Fire detection System - Raceway & Equipment Layout - Reactor Auxiliary Building El. -35.00'	0
G-FP-0019	Fire detection System - Raceway & Equipment Layout - Reactor Auxiliary Building El. -4.00'	0
G-FP-0021	Fire detection System - Raceway & Equipment Layout - Reactor Auxiliary Building El. +21.00'	0

Number	Title	Revision
G-FP-0022	Fire detection System - Raceway & Equipment Layout - Reactor Auxiliary Building El. +21.00'	0
G-FP-0023	Fire detection System - Raceway & Equipment Layout - Reactor Auxiliary Building El. +35.00'	1
G-FP-0025	Fire detection System - Raceway & Equipment Layout - Reactor Auxiliary Building El. +46.00'	0
LOU-1564, B-289 Sheets 70 & 71	Power Distribution and Motor Data 480V MCC 3A312-S One Line Diagram	21
LOU-1564, B-289 Sheet 72	Power Distribution and Motor Data 480V MCC 3A312-S One Line Diagram	10
LOU-1564, B-289 Sheet 73	Power Distribution and Motor Data 480V MCC 3312-S One Line Diagram	20
LOU-1564, B-289 Sheet 74	Power Distribution and Motor Data 480V MCC 3B312-S One Line Diagram	15
LOU-1564, B-289 Sheet 75	Power Distribution and Motor Data 480V MCC 3B312-S One Line Diagram	13
LOU-1564, B-289 Sheet 76	Power Distribution and Motor Data 480V MCC 3B312-S One Line Diagram	16
LOU-1564, B-289 Sheet 120A	Power Distribution and Motor Data 120/208V Fuse Panel FP-360-SA	5
LOU-1564, B-289 Sheet 121A	Power Distribution and Motor Data 120/208V Fuse Panel FP-361-SB	6
LOU-1564, B-289 Sheet 143	Power Distribution & Motor Data 120 Distribution Panel No. 3MA-S	12
LOU-1564, B-289 Sheet 144	Power Distribution & Motor Data 120 Distribution Panel No. 3MB-S	12
LOU-1564, B-289 Sheet 145	Power Distribution & Motor Data 120 Distribution Panel No. 3MC-S	8
LOU-1564, B-289 Sheet 146	Power Distribution & Motor Data 120 Distribution Panel No. 3MD-S	12
LOU-1564, B-289 Sheet 212	Power Distribution & Motor Data 120/208V Distribution Panel No. 3004	9
LOU-1564, B-289 Sheet 213	Power Distribution & Motor Data 120/208V Distribution Panel No. 3005	10

Number	Title	Revision
LOU-1564, B-424 Sheet 267	Control Wiring Diagram - Pressurizer Pressure (Wide Range) Sh. 4	11
LOU-1564, B-424 Sheet 268s	Control Wiring Diagram - Pressurizer Pressure (Wide Range) Sh. 5	9
LOU-1564, B-424 Sheet 269s	Control Wiring Diagram - Pressurizer Pressure Sh. 6	14
LOU-1564, B-424 Sheet 270s	Control Wiring Diagram - Pressurizer Pressure Sh. 7	16
LOU-1564, B-424 Sheet 932S	Control Wiring Diagram - Reactor Coolant Vent Valves Sh. 1	13
LOU-1564, B-424 Sheet 933S	Control Wiring Diagram - Reactor Coolant Vent Valves Sh. 2	17
LOU-1564, B-424 Sheet 934S	Control Wiring Diagram - Reactor Coolant Vent Valves Sh. 3	13
LOU-1564, B-424 Sheet 935S	Control Wiring Diagram - Reactor Coolant Vent Valves Sh. 4	8
LOU-1564, B-424 Sheet 2937	Control Wiring Diagram - Isolation Switch Device - Auxiliary Panel 1B	2
LOU-1564, B-424 Sheet 2938	Control Wiring Diagram - Isolation Switch Device - Auxiliary Panel 4A	3
LOU-1564, B-424 Sheet 2939	Control Wiring Diagram - Isolation Switch Device - Auxiliary Panel 2B	3
LOU-1564, B-424 Sheet 2940	Control Wiring Diagram - Transfer Switches Annunciation	9
LOU-1564, B-424 Sheet 2941	Control Wiring Diagram - Transfer Switch Development Auxiliary Panel - 1	11
LOU-1564, B-424 Sheet 2942	Control Wiring Diagram - Transfer Switch Development Auxiliary Panel - 2	13
LOU-1564, B-424 Sheet 2943	Control Wiring Diagram - Transfer Switch Development Auxiliary Panel - 3	5
LOU-1564, B-424 Sheet E1646	Control Wiring Diagram - Steam Line 1 Isolation Valve 2MS-V602A Sh.1	5
LOU-1564, B-424 Sheet 1646S	Control Wiring Diagram - Steam Line 1 Isolation Valve MS-V602A Sh.1	15

Number	Title	Revision
LOU-1564, B-424 Sheet E1647S	Control Wiring Diagram - Steam Line 1 Isolation Valve 2MS-V602A Sh. 2	19
LOU-1564, B-424 Sheet A1647	Control Wiring Diagram - Steam Line 1 Isolation Valve 2MS-V602A Hydraulic System	5
LOU-1564-G357	Control Room Lighting	22
LOU-1564-G358	Turbine Generator Building Ground Floor - Lighting	12
LOU-1564-G359	Turbine Generator Building Mezzanine Floor - Lighting	13
LOU-1564-G362	Reactor Containment Building Lighting - Sheet 1	8
LOU-1564-G363	Reactor Containment Building Lighting - Sheet 2	8
LOU-1564-G364	Cable Vault and Electrical Equipment Room Lighting	22
LOU-1564-G365	Reactor Aux. Building Lighting - Sheet 1	13
LOU-1564-G366	Reactor Aux. Building Lighting - Sheet 2	18
LOU-1564-G367	Reactor Aux. Building Lighting - Sheet 3	13
LOU-1564-G371	Reactor Aux. Building Lighting - Sheet 4	12
LOU-1564-G375	Cooling Tower Area Lighting	7

MAINTENANCE ACTION ITEMS

Number	Description	Date
418204	Emergency Safe Shutdown 8 Hour Battery Pack	August 9, 2000
434717	Emergency Lighting Units Float Full Battery Conditioning Cycle Test	August 14, 2002
434728	Emergency Lighting Units Float Full Battery Conditioning Cycle Test	October 16, 2002
435360	Emergency Lighting Units Float Full Battery Conditioning Cycle Test	August 15, 2002
437011	Emergency Lighting Units Float Full Battery Conditioning Cycle Test	September 10, 2002

PROCEDURES

Number	Title	Revision
DC-121	Maintenance Rule	1
ME-004-445	Self Contained Battery Powered Emergency Lighting Units	14(1)
NTP-101	Operations Training Procedure	18
OP-901-120	Pressurizer Pressure Control Malfunction	2
OP-901-502	Evacuation of Control Room and Subsequent Plant Shutdown	8
OP-901-503	Isolation Panel Fire	2
OP-901-524	Fire in Areas Affecting Safe Shutdown	0
OP-902-000	Standard Post Trip Actions	9
OP-903-126	Functional Testing of LCP-43	1
UNT-005-013	Fire Protection Program	8
UNT-007-019	Heat Stress program	5

MISCELLANEOUS

Letter, W3F-2003-007, February 25, 2003, Licensee to NRC Document Control Desk, Appendix R Deviation Request Supplemental Information (non-isolation of control circuit neutral wires between control room and Remote Shutdown Panel {for indicating lamps}, Engineering Evaluation ER-W3-2000-0817-001-00)

Letter, September 11, 2003, NRC to Licensee, Request for Deviation from Requirements of Appendix R to Part 50 of 10CFR (TAC MB3724) (3.G separation requirements for 3 cables above Refueling Water Storage Pool- found acceptable)

Letter, August 17, 1998, NRC to Licensee, Request for Exemption from Requirements of Appendix R to Part 50 of 10CFR (TAC M98144), (installation of lube oil collection system under RCP oil fill tube).

Letter, January 17, 1998, NRC to Licensee, Request for Re-evaluation of Exemption from 10CFR50, App. R (TAC M80691), (1.no fire dampers in RAB vestibule, 2. No automatic suppression in vestibule, between 8A & 8B.- found acceptable)

February 7, 1989, NRC to Licensee, Issuance of Amendment No.50 to Facility License (changes App. A Technical Specifications and License condition 2.C.9 in response to GLs 86-10 & 88-12, as approved in SER through Supplement 9 & Environmental Protection Plan in Appendix B)

Letter, February 19, 1987, NRC to Licensee, Issuance of Amendment No.15 to Facility License (changes App. A Technical Specifications by adding smoke detectors in control room and requirements for fire detection in the annulus)

Fire System Impairment List for August 11 and 25, 2003

Vendor Technical Manual for Dual Lite AC Emergency Power Systems, TM-D996.0005 457000459, Revision 13

Waterford Steam Electric Station, Unit 3 - NRC Inspection Rerport No. 50-382/00-07, dated January 31, 2001

Classroom Training Materials for Procedure OP-901-502 Training, WLP-OPS-PP051.PPT, February 25, 2003

Off-Normal Procedures OP-901-502 and OP-901-523 Lesson Plan, WLP-OPS-PPO51, Revision 2

Cable and Raceway Schedule

Waterford 3 Steam Electric Station Updated Safety Analysis Report, Revision 11-A, dated February 2, 2002

Change Package 06513, "Thermo-Lag Removed for A-16 Fire Stop," Revision 0

Change Package 06605, "Install Manual Controller for ARVs in Fire Area A-23 in Auxiliary Building," Revision 9

Technical Specifications (Appendix "A" to Facility Operating License No. NPF-38) for the Waterford 3 Steam Electric Station

Facility Operating License NPF-38 for Waterford 3.

Plant Modification ER-W3-2001-1024-000, "Appendix R Modifications"

NUREG-0787, "Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit 3, dated July 1981

NUREG-0787, "Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit 3, Supplement 3, dated April 1982

NUREG-0787, "Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit 3, Supplement 5, dated June 1983

NUREG-0787, "Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit 3, Supplement 6, dated June 1984

NUREG-0787, "Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit 3, Supplement 8, dated December 1984

NUREG-0787, "Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit 3, Supplement 10, dated March 1985

ATTACHMENT 2

White Paper Responses To Items Identified

During Waterford 3 Triennial Fire Protection Inspection

Conducted

August 11, 2003 through December 18, 2003

September 15, 2003

White Paper Responses To Items Identified During Waterford 3 Triennial Fire Protection Inspection Conducted August 11, 2003 through August 29, 2003

SECTION I: Purpose

This white paper was prepared to provide the NRC with preliminary responses to the items identified by the inspection team during the Waterford 3 Triennial Fire Protection Inspection conducted August 11, 2003 through August 29, 2003. The items were presented at the briefing held at the plant on Friday, August 29, 2003.

SECTION II: Executive Summary

Item Number: 1

NRC Concern

There is an apparent lack of documentation which provides objective evidence that OP-901-502, Control Room Evacuation Procedure, has been validated against the Safe Shutdown Analysis for time critical manual actions.

Waterford 3's Perspective

- OP-901-502 was issued in accordance with the Waterford 3 procedure review and approval process. This process includes an objective review performed by Operations and Engineering personnel to ensure that the manual actions are achievable.
- Critical aspects of plant operation have been tested using the simulator during training cycles (although not exclusively for various Appendix R scenarios).
- Documentation of OP-901-502 validation (through step 17) is provided in Attachment 3. This validation walk down of the procedure was performed on July 22, 2003. Steps 1 through 17 of the procedure encompass the most time critical manual actions (within 25 minutes).
- The Control Room evacuation drill, conducted during the August 2003 inspection, provides assurance that Waterford 3 can perform the time-critical actions noted in calculation EC-F00-026, Post Fire Safe Shutdown Analysis.

Entergy agrees with the NRC that enhancements can be made to the Control Room Evacuation Procedure. See Section III for more details on enhancements that have already been made and that are being evaluated.

Safety Significance

There is no safety significance associated with this item. We have demonstrated through training and drill performance the ability to achieve and maintain safe shutdown. The guiding principle of the Waterford 3 SSA is to perform those manual actions necessary to safely shutdown the plant and those manual actions necessary to address spurious equipment operation.

Item Number: 2

NRC Concern

a. Specific manual action locations have inadequate lighting.

b. Emergency lighting failure rates may be high enough to affect their ability to perform their function and cause the system to enter Maintenance Rule a(1) status.

c. Emergency lighting illumination test was not completed.

d. Commitment to document lighted pathways was not carried out.

Waterford 3's Perspective

a. Entergy agrees that specific identified locations requiring emergency lighting have inadequate lighting. This is being addressed in the Corrective Action Program under Condition Report CR-WF3-2003-02286.

b. Entergy is reevaluating the Maintenance Rule criteria currently applied to the Emergency Lighting System to determine if the individual failure rates are indicative of less than desirable component reliability. This evaluation is being conducted under the Corrective Action Program (Condition Report CR-WF3-2003-2439).

c. Entergy has reviewed the test cancellation and agrees with the need for additional testing. Entergy will provide the results of this test to the NRC. This issue is being tracked in the Corrective Action Program under CR-WF3-2003-02455 and CR-WF3-2003-2445.

d. Entergy does not understand the NRC concern on this item. The commitment (A-24525) was revised using the Waterford 3 commitment change process. This process includes a procedural step to notify the NRC of the change, which was accomplished via telephone. A copy of the commitment management change documentation containing the appropriate internal approvals is provided in Attachment 5.

Safety Significance

There is no safety significance associated with this item. This is based on compensatory measures that were in place (Operators carrying flashlights), isolated individual light unit abnormalities, and the administrative nature of the issues.

Item Number: 3

NRC Concern

a. Generic Letter 86-10 requires an exemption for actions, other than a reactor trip, performed prior to Control Room evacuation.

b. Actions taken prior to Control Room evacuation require verification.

Waterford 3's Perspective

a. Entergy does not understand this NRC concern. After a review of Generic Letter 86-10, Waterford 3 was unable to find a requirement for such an exemption.

b. Actions taken prior to Control Room evacuation are procedurally verified within 10 minutes of Control Room evacuation as dictated by plant conditions, either directly or indirectly through subsequent manual actions. See Section III of this paper for more details as to how this is accomplished for each action taken by the Operator prior to evacuating the Control Room.

Safety Significance

There is no safety significance associated with this item since NRC guidance documents do not require an exemption for crediting actions taken prior to Control Room evacuation and since the actions are verified in a timely manner.

Item Number: 4

NRC Concern

The reactor head vents were not evaluated as an Appendix R high-low pressure interface.

Waterford 3's Perspective

Entergy has reviewed this issue, and the reactor head vents had been previously evaluated as a high to low pressure interface as detailed in FSAR section 9.5.1.4.3 (Attachment 4). The safe shutdown analysis also evaluated the reactor head vents as a high to low pressure interface. The potential of a fire induced LOCA is mitigated by the ability to limit coolant loss to within the capability of a single charging pump by way of a flow limiting orifice.

Safety Significance

There is no safety significance associated with this item since the head vents have been previously evaluated and the physical arrangement precludes a fire induced LOCA.

Item Number: 5

NRC Concern

A manual action in Fire Area RAB-7B requires re-entry into the area within 10 minutes of a fire event in Fire Area RAB-7A and smoke levels may impair the operator action.

Waterford 3's Perspective

The required Operator action is to operate switch FR-4, in Aux Panel 2B. This action has been evaluated and determined to be feasible within the established timeframe (Calculation EC-F00-0026 Revision 1 dated 10/16/02, Post-Fire Safe Shutdown Analysis). Section III of this paper provides additional details as to how this action is accomplished.

Safety Significance

There is no safety significance associated with this item since the manual action is feasible.

Item Number: 6

NRC Concern

The FR-2 fire switch contacts in the mini-sequencer circuits were not periodically tested by plant procedures. This switch acts, in conjunction with the mini-sequencer, to automatically start selected components on the Emergency Diesel Generator B.

Waterford 3's Perspective

Entergy agrees with the NRC concern on this item. Currently, Waterford 3 does not have a program to test switch FR-2. Waterford 3 intends to initiate a program for FR switch testing.

Safety Significance

There is no safety significance associated with this item. In the remote event that the FR-2 switch has to be operated and a contact does not result in circuit continuity, operator actions provided in the existing procedure, OP-901-502 "Evacuation of the Control Room and Subsequent Plant Shutdown," will ensure operation of the components.

SECTION III: Further Details on Items

Item Number: 1

NRC CONCERN

There is an apparent lack of documentation which provides objective evidence that OP-901-502, Control Room Evacuation Procedure, has been validated against the Safe Shutdown Analysis for time critical manual actions.

Waterford 3's Perspective:

1. Documentation of OP-901-502 validation has been provided. This document, which is attached and titled "Time Validation", provides validation for actions required in 25 minutes or less. Documented times are well within requirements listed in the safety analysis. Individual step time requirements were not documented for OP-901-502 during procedure development, but walkdowns performed by procedure reviewers, followed by training, ensured the procedure could be performed within requirements.

2. The Control Room evacuation drill, conducted during the recent inspection, supports performance of time critical manual actions being within requirements. This drill scenario was provided by the inspection team and utilized Waterford 3 QA personnel to document time results for step performance. When accounting for delays that would have not occurred during an actual evacuation of control room event, and incorrect cues and indications provided at LCP-43 (Remote Shutdown Panel), all time requirements were met.

   Attachments titled "8-26-2003 Appendix R Drill" and "Comparison of 8-26-2003 Drill Performance to Requirements" documents the drill control issues, reasons for delays, and the timed completion of each step performance.

3. Waterford 3 will assess the industry for best practices related to procedure verification and validation processes (Reference CR-WF3-2003-2440). The current version of OP-901-502 was validated as follows:

   - The validation document provided to the inspection team is based upon a Licensed Operator walkdown and review by a Senior Licensed Operator, of the simulated performance of actions as listed in OP-901-502. This documentation method is consistent with previous inspection team requests and was satisfactorily received during the previous inspection.

- The revision to OP-901-502 was performed and reviewed utilizing the 10CFR50.59 review process (i.e. 50.59 screen). An objective review performed by operations and engineering personnel, supports that procedure performance and all time critical manual actions are within safety analysis requirements. This review consisted of a walkdown by operations personnel to ensure time requirements could objectively be met.

Additionally, Waterford 3 agrees there are enhancements which will support the goals of OP-901-502. CR-WF3-2003-02440 has been generated to review and evaluate appropriate actions for this item.

The following are some enhancements that have been made to the procedure or that are being evaluated:

- Requirements for time critical actions have been incorporated in OP-901-502 under Revision 8 Change 3. Further evaluations will be done to determine if placement into an alternate controlled method is utilized (Tech Guide, Training Lesson Plan, etc). Change 3 added notes prior to step numbers and an attachment, to be used as a quick reference, which provides information concerning time critical steps along with system parameters which may indicate step performance is required.

- Waterford 3 is reviewing all comments made by the inspection team as well as information obtained from critiques held by the operation's shift and QA personnel which supported the drill. These comments will be evaluated along with other feedback obtained from the industry to improve OP-901-502 and any other related documents.

Safety Significance:

There is no safety significance associated with this item. We have demonstrated through training and drill performance the ability to achieve and maintain safe shutdown.

Item Number: 2

NRC Concern:

The following items were identified associated with Appendix R, 8-hour emergency lighting.

a. Specific manual action locations have inadequate lighting.

b. Emergency lighting failure rates may be high enough to affect their ability to perform their function and cause the system to enter Maintenance Rule a(1) status.

c. Emergency lighting illumination test not completed.

d. Commitment to document lighted pathways not completed.

Waterford 3's Perspective:

a). Preliminary indication is a minimum of two locations where Operations performs manual actions associated with OP-901-502 require additional lighting units. Identified emergency lighting discrepancies are being addressed in the Corrective Action Program under Condition Report CR-WF3-2003-02286.

b). The Emergency Light system (LTE) is presently included in the Maintenance Rule (MRule) as described under 10CFR50.65. Present MRule criteria calls for an availability rate of >90% for the total light unit population. The condition identified involved small sample sets of light units subjected to periodic testing. These tests yielded a higher failure rate than the MRule criteria when taken as an individual test, but within the acceptable criteria when compared to the total LTE population. Condition Report CR-WF3-2003-2439 was generated to ensure the MRule criteria currently applied to the LTE system are appropriate to ensure component reliability.

c). Considering the system deficiencies noted during the inspection, it is clear that performing the scheduled test would have identified these deficiencies. Test cancellation is being reviewed internally. This review is being tracked in the Corrective Action Program under CR-WF3-2003-02455.

d). A copy of the associated Commitment Management Change documentation has been provided to the inspection staff which details the process and approvals completed in changing this commitment. The staff indicated this information had not been reviewed during the site visit and would be considered prior to the final report.

Actions to Address the Item(s):

Immediate action was taken by Operations management to verify that flashlights were staged in the Control Room for use during an Appendix R fire. The Operator shifts have been briefed on this issue.

Additional light units will be added as necessary under ER-W3-2003-00563-000, which was generated for this purpose. In the interim compensatory actions are in place which include the staging of supplemental flashlights and batteries in the Control Room for use when performing manual actions. This is in addition to the flashlights provided in the Appendix R Equipment Lockers located in the RAB +35 elev. Relay Room, and the RAB +21 elev. Remote Shutdown Panel (LCP-43) Room. The emergency lighting inadequacies are being evaluated and corrective actions are being tracked under CR-WF3-2003-02286 and CR-WF3-2003-02445.

Evaluation of the cancelled test is being performed under the Corrective Action Program under Condition Report CR-WF3-2003-02455. Waterford 3 will conduct the previously scheduled test.

Safety Significance:

There is no safety significance associated with this item. It is standard practice for Operators to carry and/or have access to flashlights while performing duties in the plant, this is also detailed in OP-100-001, Operations Standards and Management Expectations. Likewise, compensatory measures associated with emergency lighting is for the Operators to carry and/or have access to flashlights while performing licensed duties in the plant. While Appendix R does require 8-hour battery powered lights, the Operators are able to perform safe shutdown manual actions using these compensatory measures in the areas where deficient light levels were identified.

Item Number: 3

NRC Concern:

(a) Generic Letter 86-10 requires an exemption for any actions, other than a reactor trip, performed prior to evacuating the Control Room
(b) Actions performed prior to evacuating the Control Room require verification.

Waterford 3's Perspective:

(a) Entergy's review of Generic Letter 86-10 did not identify a requirement for an exemption for actions performed prior to evacuation of the Control Room. Statements relative to requiring an exemption were provided in the Fire Protection Policy Steering Committee Report (Generic Letter 85-01). GL85-01 was a draft document issued for comments and was later formally issued as GL86-10. Question 3.8.4 "Control Room Fire Considerations" was revised upon issuance of GL86-10 deleting the statement requiring an exemption.

(b) Actions taken prior to Control Room evacuation are procedurally verified within 10 minutes of Control Room evacuation as dictated by plant conditions, either directly or indirectly through subsequent manual actions. There are six actions prescribed in OP-901-502, "*Evacuation of Control Room and Subsequent Plant Shutdown*", in addition to tripping the reactor that the Operators perform as a result of a Control Room/Cable Vault fire. These actions and their verification activity is discussed below:

1. Close MS-124A and MS-124B (Main Steam Isolation Valves)

   OP-901-502 contains actions to operate fire isolation switches FR-3 and FR-4 at Auxiliary Panel 2B. These actions isolate the Control Room circuits and align the Train 'B' DC power supply (circuit 10 at DC-EPDP-1B-DC) directly to the Train 'B' powered solenoids, which ensures that the respective MSIV is energized and closed. The confirmatory actions can be completed within 10 minutes of control room evacuation and are appropriate and timely actions.

2. Place MS-116A and MS-116B in Manual

   MS-116A and MS-116B are the atmospheric Dump valves (ADVs). Timely actions are only required if an ADV were to spuriously open. This procedural step can be completed within 10 minutes of a spurious opening of the ADV. Indication of a spurious opening is provided at LCP-43 by monitoring plant parameters. OP-901-502 contains actions for manual closure of an ADV.

3. Place all Charging Pumps in Off

   The safe shutdown analysis states (Appendix D section 5.2) that the spurious closure of the volume control tank isolation valve (VCT-183) could result in the loss of head to any running charging pumps and possible subsequent pump damage. Since it is not known which charging pump may be operating at the time of fire, the control room action to place the charging pumps in OFF is a conservative action pending verification that a suction path is provided from the boric acid makeup tanks (BAMT) or Refueling Water Storage Pool (RWSP). Spurious operation of the non-credited charging pumps are terminated at the remote shutdown panel, LCP43, (if necessary) following operation of transfer switches at Auxiliary Panel 3 (charging pump AB), transfer switches at Auxiliary Panel 1 (charging pump A), and the placement of isolation switch SS/377 at local control panel LCP-80SB to the ISOLATE position (charging pump B). In addition, should the fuse fail prior to transfer of charging pump B (the credited pump for the Control Room/Cable Vault fire) the procedure allows operation of the charging pump from the breaker for hot shutdown and replacement of the fuse to support cold shutdown RCS inventory and pressure control (breaker operation is provided as a means to operate the pump since manual fuse replacement is not allowed for hot shutdown actions). The isolation actions are in effect verification actions, and can be performed within 10 minutes of control room evacuation. These actions are appropriate and timely.

4. Place the normal Spray Valves in “BOTH”.

   This step requires timely verification since another step taken by the Operator, prior to leaving the Control Room, (de-energizing the 7kv bus) will take power off of the Reactor Coolant Pumps (disabling normal spray path). See #5 below.

5. Trip the Reactor Coolant Pumps.

   Actions are taken in the Control Room to ensure that the spray valves (RC-301A and RC-301B) are closed and to trip the Reactor Coolant Pumps. Should circuit damage preclude these actions, confirmatory Operator actions are provided in the Turbine Building to trip the 7kv buses feeding the RCPs. These confirmatory actions can be completed within 10 minutes of a spurious Spray Valve actuation. These actions are appropriate and timely.

6. Isolate Letdown by closing CVC-101 and CVC-103.

   This control room action closes both valves CVC-101 and CVC-103. Only one valve is required to be closed to isolate letdown. Thus it would require two separate spurious conditions to fail both valves in the open position. A confirmatory indication that confirms CVC-103 closed is provided at the remote shutdown panel (LCP-43). This action can be performed within 10 minutes of Control Room evacuation and is appropriate and timely. The procedure provided verification of the Control Room actions through subsequent actions. Additional verification, based on NRC inspection team comments, has been added to the procedure. Procedure step 5.1 has been added to provide further verification of valve closure.

Actions to Address the Item:

Step 5.1 has been added to provide further Operator direction for verification of closure of CVC-101 and CVC-103 (isolating letdown). This is considered an enhancement to the procedure. The step states to verify that either of the valves have closed.

Step 5.2 has been added to provide further operator direction for verification that the Charging Pumps are secured. The step states to “verify all charging pumps are secured”. This is considered a procedure enhancement.

Condition Report CR-WF3-2003-02448 has been generated to review and evaluate any further appropriate actions for this item.

Safety Significance:

There is no safety significance associated with this item because timely verification actions were prescribed in the procedure.

Item Number: 4

NRC Concern:

The NRC was concerned that the reactor head vents had not been considered as an Appendix R high-low pressure interface.

Waterford 3's Perspective:

The reactor head vents have been previously evaluated as a high to low pressure interface as detailed in FSAR section 9.5.1.4.3. A copy of the FSAR section has been provided to the inspection team. In addition the safe shutdown analysis also evaluated the reactor head vents as a high to low pressure interface. However the discussion in the safe shutdown analysis is fragmented and requires additional clarity.
The potential of a fire induced LOCA is mitigated by the ability to limit coolant loss to within the capability of a single Charging Pump by way of a flow limiting orifice.

The issue related to the reactor head vent valves is derived from concerns identified in Generic Letters 81-12, 86-10 and even more recently in Reg. Guide 1.189. Generic Letter 81-12 provides the details and states that a "..single fire could cause two valves to open resulting in a fire-initiated LOCA through the subject high-low pressure system interface" GL 81-12 provides the following guidance in regards to addressing high-low pressure interfaces:

A. Identify each high-low pressure interface that uses redundant electrically controlled devices (such as two series motor operated valves) to isolate or preclude rupture of any primary coolant boundary.

B. Identify the device's essential cabling (power and control) and describe the cable routing (by fire area) from source to termination.

C. Identify each location where the identified cables are separated by less than a wall having a three-hour fire rating from cables for the redundant device.

D. For the areas identified in item 2.C above (if any), provide the bases and justification as to the acceptability of the existing design or any proposed modifications."

The Waterford 3 safe shutdown analysis (Calculation EC-F00-026, Section 5.3.15 titled "High Pressure to Low Pressure Interface") evaluated the Reactor

Coolant Head Vent System as a high pressure to low pressure interface. The analysis states:

"As stated in 10CFR50.46 and 10CFR50 Appendix A, Definitions, Loss-of-Coolant Accidents (LOCAs) are the hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the largest pipe in the reactor coolant system. Although fires are not postulated to cause a pipe rupture (Section 5.2.1.2), fire induced cable damage may cause spurious operation of components relied upon to maintain the integrity of the reactor coolant pressure boundary (RCPB). The loss of this RCPB integrity due to fire induced spurious operation of one or more components shall not result in a loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system."

The valves of concern in the Reactor Head Vent System are solenoid valves RC-1014, RC-1015, RC-1017 and RC-3186. The solenoid valves are "energize to open" and "fail close" upon loss of power. These 1-inch valves are normally closed with key removable switches normally in the "OFF" position. There are two key switches in the Control Room which provide power to the individual valve control switches. One keyed power switch is for the "A" train valve controllers and one keyed power switch is for the "B" train valve controllers. RC-1015 and RC-3186 are powered from the "A" train power key switch and associated valve controllers. RC-1014 and RC-1017 are powered from the "B" train power key switch and associated valve controllers. Each switch is provided with a separate cable. Multiple cable to cable fire induced interactions are required to cause two valves to spuriously open establishing a flow path to the Quench Tank. However assuming these valves open, an orifice is provided in the line limiting the flow to within the capability of a single charging pump. Thus the performance goal (10CFR50.46 and 10CFR50 Appendix A) of reactor inventory loss within the capability of the makeup system has been satisfied.

The safe shutdown analysis compliance summary section for Fire Area RAB 1 (Control Room/Cable Vault) provides the cable by cable analysis for the cables associated with the above listed valves. The compliance statement for RC-1014 and RC-1015 states "RC-1017 and RC-3186 remain available through local manual action to provide reactor head vent isolation." The compliance statement for RC-1017 states "Operation of FR-5 Switch at Auxiliary Panel 2B in Fire Area RAB7B will de-energize RC-1017 and fail the valve in the required closed position". The compliance statement for RC-3186 states "Operation of FR-1 Switch at Auxiliary Panel 1B in Fire Area RAB7A will de-energize RC-3186 and fail the valve in the required closed position."

These actions are provided for inventory control purposes because the performance goal of maintaining the loss of reactor coolant within the capability of the makeup system has been met. As indicated above, the safe shutdown analysis has evaluated this high-low interface but it is apparent that the document lacks the necessary clarity.

Actions to Address the Item:

This item has been previously evaluated as documented in FSAR Section 9.5.1.4.3 and the safe shutdown analysis. The safe shutdown analysis will be revised to provide an enhanced level of clarity.

Condition Report CR-WF3-2003-02451 was generated to review and evaluate appropriate actions for this item.

Safety Significance:

There is no safety significance associated with this item, since the subject condition has been evaluated in the FSAR and the Safe Shutdown Analysis and the physical arrangement precludes a fire induced LOCA.

Item Number: 5

NRC Concern:

A manual action in Fire Area RAB-7B requires re-entry into the area within 10 minutes of a fire event in Fire Area RAB-7A and that smoke levels may impair the operator action.

Waterford 3's Perspective:

The required action is operating FR-4 switch, in Aux Panel 2B. This action has been determined to be feasible by calculation EC-F00-0026, Post-Fire Safe Shutdown Analysis. The manual action will be accomplished by either the on-shift operators, or the fire brigade (note that the fire brigade consists of only operators) as directed by the Control Room.

The relay room contains a very low combustible loading and is provided with both smoke detection and sprinkler protection. Smoke detection provides early response while smoke levels are minimal and sprinkler protection further limits the size of the fire, and amount of smoke generated. Additionally the fire brigade utilizes smoke ejector fans to establish ventilation during the firefighting operations. These fans have the capacity to accomplish an entire room air change in less than 4 minutes, and are located in each of three fire lockers (the closest being on +46 Elev. RAB, just up the stairs from the Relay Room). Also note that the fire brigade is provided with full protective clothing and self contained breathing apparatus (SCBA) and would be physically protected from any fire or fire products present.

The manual action ensures closure of the MS-124 valves (Main Steam Isolation Valves). This is a credited action and one chosen as the boundary for Appendix R purposes. Failure to perform this action is only a concern in the event of additional concurrent failures.

SSER #8, in the section 9.5.1.3, (page discussing the deviation for partial height walls) states,

> "The fire load in the area on either side of the walls is low."
>
> "The walls themselves are constructed of materials that possess at least a 1 hour fire rating. Therefore, they will be able to provide protection from the effects of direct flame impingement and radiant heat from a fire. Although the walls will not prevent hot gases from passing from one area to another over the top of the barrier, the components of concern are located at the floor level and would not be affected by a hot gas layer located at the ceiling."
>
> "The areas on either side of the partial-height walls have been provided with complete smoke detection systems and automatic fire suppression systems. The staff, therefore, has reasonable assurance that any fire would be detected early and suppressed manually by the plant fire brigade or extinguished automatically by the fixed fire

suppression systems. The staff, therefore, concludes that the partial-height walls delineated in the applicant's letter of March 26, 1984, represent an acceptable deviation from the technical requirements of Section III.G.2 of Appendix R to 10CFR50."

The SSER indicates the area has been reviewed previously, no adverse affects identified, and found acceptable with respect to equipment and facility exposure to the fire conditions expected. These same conditions would likewise pose no adverse affects to fire brigade and operations personnel making entry into the area to perform the manual action.

Thus, the action is considered feasible within the time required based on the following:

1) A low probability of actions being necessary due to the concurrent failures required.
2) Early detection and suppression capability provided through defense-in-depth minimizing the size and effects of the fire.
3) Low fire loading, yielding a short duration fire of limited magnitude.
4) Ability to eliminate smoke and heat rapidly from the area.
5) Personal protection of the fire brigade allowing for their entry into the area.

Actions to Address the Item:

No additional actions are necessary for this item. This item is being evaluated in the Waterford 3 Corrective Action program under Condition Report CR-WF3-2003-02446.

Safety Significance:

There is no safety significance associated with this item since the manual action is feasible.

Item Number: 6

NRC Concern:

The FR-2 fire switch contacts in the mini-sequencer circuits were not periodically tested by plant procedures. This switch acts, in conjunction with the mini-sequencer, to automatically start selected components on the Emergency Diesel Generator B.

Waterford 3's Perspective:

During the inspection, the FR-2 switch contacts were specifically questioned. Subsequent reviews by Waterford 3 Engineering indicate that other FR Switches (FR-1, FR-3, FR-4, and -5) are also not tested by plant procedures. Transfer switches designated as "43" are operated during a control room evacuation. These "43" switches are periodically tested using plant procedure OP-903-126 "Functional Testing of LCP-43".

The mini-sequencer relays are bench tested by preventive maintenance procedures every 3 years. The last tests on these relays were performed under MAI #'s 419606, 419608, 419609, and 419611. In addition, procedure OP-903-116 "Train B Integrated EDG/ESFAS" also verifies the mini-sequencer relay operation every 18 months.

The FR switches are part of the plant equipment used to mitigate the consequences of a fire in the Control Room or the Cable Spreading Room. Fire protection is addressed in TRM 3.3.3.8.1, 3.7.10 and 3.7.11. These sections of the TRM only address fire detection, fire suppression, and fire rated assemblies respectively. Additional equipment required to mitigate the consequences of a fire is specified in TRM 3.1.2.4, 3.7.3, and 3.7.12. These sections address Charging Pumps, Component Cooling Water pumps, and Essential Chilled Water only. Fire switches are not explicitly addressed in TS/TRM.

Generic Letter 96-01 required all technical specification related logic circuits to be tested. Subsequent clarification by the NRC (letter from NRC – Bruce A. Boger dated March 27, 1996) indicated that the Appendix R circuits are beyond the scope of Generic Letter 96-01. Therefore, the FR switch contacts were not required to be tested per this Generic Letter.

There is a high level of confidence that the FR contacts associated with these switches will function as:

1. The FR switch is a manual (mechanical) rotary switch.
2. There is no 'failure to operate' history associated with other similar switches (43 transfer switch) at Waterford 3.

Testing the FR switches introduces high risk of plant perturbation. For example, actuating the FR switches will result in isolating the 4160 3B Safety Bus from offsite power, starting Emergency Diesel Generator B, and automatically starting various safe shutdown loads. Therefore, planning, scheduling and testing these FR switches is a complex task.

Actions to Address the Item:

Waterford 3 intends to initiate a program for FR switch testing. The plant will determine the appropriate time (such as Safety B Bus Outage) for testing of these FR switches, to minimize the potential for challenging plant operation. Condition Report CR-WF3-2003-02441 has been generated to review and evaluate appropriate actions for this item.

Safety Significance:

There is no safety significance associated with this item. The loads associated with the FR-2 switch contacts on the mini-sequencer relay circuits are as follows:

1. Component Cooling Water Pump B (CWD 709)
2. Diesel Generator B Room Exhaust Fan E28 (CWD 1043)
3. Station Service Transformer 3B32 (Pressurizer Heaters Bus - CWD 2398)
4. Station Service Transformer 3B315 Feeder (Dry Cooling Tower Fans - CWD 2399)

The postulated scenario is a control room evacuation due to a fire and subsequent transfer of the plant control functions to LCP-43. The above loads are energized by the mini sequencer relay contacts in series with the FR-2 contacts during a control room fire to support the running of the Diesel Generator B and reduce operator actions.

In the remote event that any one of the contacts does not result in circuit continuity after switch operation, operator actions provided in OP-901-502 "Evacuation of the Control Room and Subsequent Plant Shutdown" will ensure proper operation of the components.

This procedure verifies that the required loads are in operation as follows:

The operation of CCW pump B is verified in step no. 24. If CCW Pump B has not started due to FR-2 switch contact failure, the operator can manually start the pump from LCP-43.

The failure of the EDG B room exhaust fan to start will be observed by the operator during performance of step no. 13 when the operator enters the EDG B room during local verification of the EDG B operation. The exhaust fan can be locally started at the 3B31 switchgear.

The proportional pressurizer heater bank 2 will be restored by step no. 30 if switchgear 32B is not already energized. The Station Service Transformer 3B32 Feeder breaker can be manually closed at the 3B3 Switchgear. The proportional pressurizer heaters are not essential for immediate operation. The safe shutdown analysis established a one-hour time line requirement to restore the pressurizer proportional bank no. 2.

The operation of dry cooling tower fans will be verified during performance of step no. 38. The Station Service Transformer 3B315 Feeder breaker can be manually operated at the 3B3 Switchgear. During a control room fire event, most loads (non-essential) are secured with the operation of the FR switches. Therefore, the heat load on the Component Cooling Water is minimal.

Based on the above discussions, significance of not testing the “make” capability of FR-2 switch contacts is not considered to adversely impact safe shutdown capability.

ATTACHMENT 1

COMPARISON OF 8/26/2003 DRILL PERFORMANCE TO REQUIREMENTS

(Comments are provided in Bold Print)

Compliance Assessment Summary

Analysis/Fire Area:	RAB1 (Includes RAB1A & RAB1E)
Analysis/Fire Area Description:	Control Room Proper / Cable Vault

The Safe Shutdown Systems and Equipment identified below are available to meet the safe shutdown performance goals of Appendix R Section III.L. This area is in compliance with Section III.G.3 and III.L of 10CFR50 Appendix R with approved exemptions.

Safe Shutdown Function	**System**
RCS Inventory and Pressure Control	Charging Pump B
Decay Heat Removal (HSD)	EFW Pump B with S/G No. 1 and 2
Decay Heat Removal (CSD)	LPSI Pump B
Essential Electrical Support	Train B (EDG)
Essential Mechanical Support	ECW Train B
	CCW Train B with Train B DCT
	ACCW Train B with Train B WCT
	HVC Train B
	HVR Train B

The manual actions required for this area are specified in OP-901-502. Actions include the operation of transfer switches and fire isolation switches designed to isolate control room circuits from affecting operation of the component from LCP-43 located in Fire Area RAB9.They are summarized here for information. Actions taken in the control room, other than tripping the reactor, will not impact the operator's ability to complete the other actions in a timely manner (See Appendix D, Functional Requirements Time Line Analysis). Backup, confirmatory actions will be taken for these actions.

Prior to exiting the Control Room

1. Trip the reactor.
2. Close MS-124A and MS-124B.
3. Place MS-116A and MS-116B in MANUAL.
4. Place all charging pumps in OFF.
5. Place the normal Spray Valves in "BOTH".
6. Trip the Reactor Coolant Pumps.
7. Isolate letdown by closing CVC-101 and CVC-103.

All Immediate Operator actions were completed prior to commencing the drill as directed by the NRC inspection team.

The following manual actions are required for this area. OP-901-502 explains the function of each transfer and isolation switch.

Hot Shutdown

1. Operate MS-116A and MS-116B locally on ROOF W and ROOF E, respectively. (If spuriously open, close within 10 minutes)

   Operator was dispatched in 6 minutes (time adjusted for waiting for initial phone call to commence transfer, which would not have been required if sound powered phones were used, and for additional stopping of transfer activities to answer another phone call that instructed us to consider ADV #1 as failed).

2. Operate all transfer switches and push to activate controls at LCP-43 in RAB9.

   Completed within 9 minutes due to reasons listed in steps 1-4.

3. Operate all transfer switches at Auxiliary Panel 2 in RAB7B, Auxiliary Panel 3 in RAB7D, and Auxiliary Panels 4 and 1 in RAB7A (Place switches in the AUX CR (transfer) position.).
   (10 minutes)

   Completed within 5 minutes due to reasons listed above.

4. Place all Fire Isolation Switches in the ISOLATE position at Auxiliary Panel 2B in RAB7B, and Auxiliary Panels 4A and 1B in RAB7A. Operation of these switches will also block any spurious DEFAS-A-AUTO and DEFAS-B-AUTO initiation signals. (10 minutes)

   Completed within 8 minutes due to time listed above, and time to walk back and turnover to CRS (evolution complicated by having to stop evolution and transit back and forth to the telephone).

5. Re-close breaker #10 at DC-EPDP-1B-DC in Fire Area RAB8B to restore power to the Train B dump valves and close MS-124A and MS-124B. (10 minutes)

   This action was not performed as it was not being necessary because the failure had not occurred. Had indications shown MS-124A or MS-124B had failed, an operator would have gone to these breakers instead of to the #1 ADV.

6. Place Key Switches SS/377, SS/703, and SS/1055 in the ISOLATE position at LCP-80 in RAB3. (10 minutes)

   Completed within 9 minutes due to reasons listed above.

7. Remove power from the RCPs. De-energize SWGR 1A and 1B in Fire Area TGB by opening the DC knife switches for breakers 7KVEBKR1A-4 and 7KVEBKR1B-4, then manually tripping breakers 7KVEBKR1A-4 and 7KVEBKR1B-4. (10 minutes)

   Completed within 19 minutes. This action is not required to be completed within a time limit for the given scenario (i.e. there was no spurious pressurizer spray flow). An operator established communication with the PNPO at LCP43 within 5 minutes of the onset of the event. Had indications of a spray valve failing Open been indicated, an operator was on station and prepared to take the actions necessary to secure the spray flow by de-energizing the Reactor Coolant Pumps.

   Appendix D to the safe shutdown analysis states:

   "Where circuit damage may preclude this action (trip RCPs from Control Room), confirmatory operator actions are provided in the Turbine Building to trip the 7kV buses feeding the RCPs. Based on engineering judgment, actions to secure spurious spray flow should be completed within 10 minutes."

   It should be noted that operators have an immediate operator action to secure ALL Reactor Coolant Pumps prior to leaving the Control Room for this event.

8. Take local control of EDG B by placing key-operated switch FR 2367 to ISOLATE in RAB15. (20 minutes)

Action to place FR 2367 to ISOLATE was completed within 28 minutes due to the cascading delays listed above and the additional delays of having to walk down to the - 4 control point (actual route would have been through the +21 MG Set Room), overcome delays experienced logging into the CAA, and then walking back up to the +21 to the EDG B Room.

There is no time limit for this action due to EDG B running. However, if EDG B had not been running, the time limit would have been one hour due to EFW Pump AB running. It should be noted that indications provided to the crew were that EDG B was running and supplying the safety bus.

Appendix D to the safe shutdown analysis states:

"The time available to establish emergency AC power is dependent upon the most time sensitive function to recover, which is the establishment of initial decay heat removal. This may require power to a motor driven EFW pump within 25 minutes to maintain level in at least one steam generator. If the steam driven pump is available, then emergency AC power will not be required for about one hour. If operation of a motor driven EFW pump is required, then AC power must be restored within about 20 minutes."

9. Manual action is required to close breaker No. 28 at DC-EPDP-3B-DC in Fire Area RAB 8B to restore power to the mini sequencer to support automatic EDG B start on loss of offsite power. (20 minutes)

Actual time was 30 minutes. Action is required to be performed only if the EDG B failed to start or tripped due to the fire event. There were no indications given by the NRC inspection team that the EDG B was not allowed to start automatically and supply the bus. Approximately 20 minutes after the EDG B was supplying the bus, the NRC inspection team informed the drill controller that the EDG B would suffer a failure to automatically start.

Assuming that the EDG B tripped at the time the NRC inspection team indicated the failure to start, the step to close 4KVEBKRB-28 was completed within 5 minutes. This time was still within the one hour required to restore electrical power with EFW Pump AB running.

10. Manual action is required to locally re-close SSD-ESWGR-31B supply breaker EBKR-3B-10 at 4KV-ESWGR-3B in Fire Area RAB8B. (20 minutes)

This was verified within 10 minutes of entering LCP-43 by the PNPO when he verified that he had EFW Pump B and CCW Pump B running. The SNPO verified this action in the field within 25 minutes.

With EFW Pump AB running, the next component required to be energized in the least amount of time is Charging Pump B within one hour.

11. Open breaker 30 at LVD-EPDP-61B in Fire Area RAB8B in order to fail HVR-502B to maximum pitch to support EDG B Room HVAC. (20 minutes)

This action provides a support function for EDG operation and is not time critical in relation to EDG failure. The time applied to this action is based on prioritization of steps for procedure preparation. The breaker is opened to verify ventilation is being provided to EDG B Room. This breaker was verified open within 27 minutes.

12. Manual action is required to re-close breaker #15 at ID-EPDP-91B in fire Area RAB 7B to restore power and close valves CC-127A, CC-127B, CC-134B and open CC-135B. (20 minutes)

This action was completed within 15 minutes. This action was delayed when the CRS stopped performing the procedure to explain to the NRC inspection team information related to the transfer switches. He also waited for the call from LCP-43 (in reality he would have been on sound powered phones and would have completed this step seven minutes earlier).

13. Remove control power and close breaker EBKR-3B-7 at 4KV-ESWGR-3B in Fire Area RAB8B to restore power to SSD-EMCC-315B. (20 minutes)

This step is placed at this point in Appendix C to the safe shutdown analysis to ensure CCW temperature control would be available to support the running Emergency Diesel Generator. This action was met during the drill.

The knife switch for SSDEBKR3B-7 was opened within 26 minutes. Indications were that all Dry Cooling Tower Fans on the 'B' Train were operating properly, Component Cooling Water (CCW) 'B' temperature was given as 87F by the monitor when asked by the

PNPO. There were no indications that this breaker was acting erratically threatening CCW temperature.

14. The auto MSIS-A initiation signal which may result from fire induced cable damage can be overridden as follows: (a) for EFW-229B by de-energizing the valve at DC-EPDC-A-DC breaker no. 23 in fire area RAB8A to fail the valve in the required open position (b) for EFW-223A by placing the Fire Isolation Switch FR-1 at Auxiliary Panel 1B in fire area RAB7B in the "ISOLATE" position. (Note that this was also accomplished in Step 2 above.) (25 minutes)

    FR-1 at Auxiliary Panel 1B was placed in ISOLATE within 10 minutes of exiting the Control Room. EFW-229B was verified open during this drill within 10 minutes of arriving at LCP-43 when the operator verified flow to both Steam Generators (it is required to be open for a flow path to SG2 through the Backup flow path as prescribed in the procedure).

    The procedure used for the drill per NRC direction was not the current approved procedure and thus did not contain the step to open the breaker to fail this valve open. The operator verified it as open by indications on LCP-43. The procedure was revised to incorporate this breaker into the procedure prior to performance of the drill.

15. The auto MSIS-B initiation signal which may result from fire induced cable damage can be overridden as follows: (a) for EFW-223B by placing the Fire Isolation Switch FR-4 at Auxiliary Panel 2B in fire area RAB7B in the "ISOLATE" position (b) for EFW-229A by placing the Fire Isolation Switch FR-5 at Auxiliary Panel 2B in fire area RAB7B in the "ISOLATE" position. (Note that this was also accomplished in Step 2 above.) (25 minutes)

    FR-4 at Auxiliary Panel 2B was placed in ISOLATE within 10 minutes of exiting the Control Room. FR-5 at Auxiliary Panel 2B was placed in ISOLATE within 10 minutes of exiting the Control Room.

16. Open breakers at ID-EPDP-91B in Fire Area RAB7B as specified in OP-901-502 (per ER-W3-98-1380-00-00). This will prevent the loss of ID-EUPS-B (which supports EFW, CCW, and plant monitoring instrumentation) due to faults on non-SSD loads. (25 minutes)

    This action was completed within 14 minutes. This action was delayed when the CRS stopped performing the procedure to explain to the NRC inspection team information regarding the FR switches. He also waited for the call from LCP-43 (in reality he would have been on sound powered phones and would have completed this step approximately seven minutes earlier).

17. De-energize MS-119A at SSD-EMCC-313A, compartment 6M, located in Fire Area RAB8A. Verify closed/manually close MS-119A in Fire Area ROOF W to isolate the SG-1 upstream drain line. (30 minutes)

This action was completed with 28 minutes. There was no time limit associated with this action since MS-119A was already closed.

Appendix D to the safe shutdown analysis states:

"These valves (MS-119A and MS-119B) are normally closed, and would only require closure following a fire-induced fault (hot short) causing the valve to spuriously open. These steam drains are 2" lines that drain to the main condenser. Inventory loss through these lines is not significant, but the valve should nonetheless be closed to fully isolate the steam generator to preserve steam generator inventory and control cool down. Engineering judgment dictates that this action should be completed within thirty minutes of the event."

18. De-energize MS-119B at SSD-EMCC-313B, compartment 6M, located in Fire Area RAB8B. Verify closed/manually close MS-119B in Fire Area ROOF E to isolate the SG-2 upstream drain line. (30 minutes)

This action was completed within 29 minutes. There was no time limit associated with this action since MS-119B was already closed.

Appendix D to the safe shutdown analysis states:

"These valves (MS-119A and MS-119B) are normally closed, and would only require closure following a fire-induced fault (hot short) causing the valve to spuriously open. These steam drains are 2" lines that drain to the main condenser. Inventory loss through these lines is not significant, but the valve should nonetheless be closed to fully isolate the steam generator to preserve steam generator inventory and control cool down. Engineering judgment dictates that this action should be completed within approximately thirty minutes of the event. "

To mitigate spurious actuation of the following components, de-energize Train A electrical power: 1) open the DC knife switch for EBKR3A-11 and trip the breaker at 4KV-ESWGR-3A in Fire Area RAB8A and 2) locally trip DGA in Fire Area RAB 16. (30 minutes)

- BD-103A (fail to required closed position)
- BD-103B (fail to required closed position)
- CS-MPMP-0001A
- EFW-MPMP-0001A
- SI-MPMP-0002A
- SI-MPMP-0002A/B (Operation of FR-2 isolates the AB busses from Train B busses)

These actions were completed within 47 minutes. This 'A' train is separated from the 'B' train, and the 'B' train is separated from the Control Room. Therefore spurious actions on the 'A' train will not affect the plants ability to safely shutdown. For the plant conditions during the drill, and expected plant conditions during an actual event (RCS pressure greater than HPSI shutoff head), the above listed spurious action with the most impact would be Containment Spray Pump A.

Appendix D of the safe shutdown analysis states:

"Inadvertent Containment Spray Initiation is a concern since the Containment Spray (CS) Pumps (CS-MPMP-0001A and CS-MPMP-0001B) take their suction from the RWSP, which is a credited source of borated makeup water required to reach cold shutdown. The rated flow rate for these pumps is 1810 gpm, and could start due to a pump control circuit hot short or a spurious SIAS or CSAS signal. The SIAS or CSAS spurious start can be secured by placing the Control Room hand switch in OFF. If the pump's control circuit is directly affected, then opening the pump breaker will be required. Due to the volume of water in the RWSP (minimum Technical Specification Level is 475,500 gallons) compared with the amount needed for plant cool down (a maximum of 58,000 gallons from the RWSP), mitigation of this spurious actuation is not a time critical function (operation of the pump for 1 hour at 1810 gpm would use 108,600 gallons). Actions to secure a spuriously started CS Pump should be completed expeditiously, but no so as to interfere with other time critical activities."

It should be noted that there were no indications of spurious operation on the 'A' train equipment during the drill.

19. De-energize SSL-8004A at ID-EPDP-91B, breaker no. 49, located in Fire Area RAB7B in order to fail the valve closed and support isolation of SG#1. (30 minutes)

    This action was completed within 14 minutes.

20. De-energize SSL-8004B at ID-EPDP-91B, breaker no. 45, located in Fire Area RAB7B in order to fail the valve closed and support isolation of SG#2. (30 minutes)

    This action was completed within 14 minutes.

21. De-energize CVC-209 at DC-EPDP-AB-DC, Circuit # 38 in fire area RAB8C to fail CVC-209 in the required open position to support charging for RCS inventory and pressure control. (1 hour)

    This action was completely within 24 minutes.

22. Place fuse selector switch to BACKUP on CVC-EBKR311B-10M in Fire Area RAB8B to restore operation of CVC-183 from LCP-43. The valve should be closed to isolate the VCT. (1 hour)

This action to operate CVC-183 was completed within 54 minutes from LCP-43. The contingency to operate the fuse selector switch was not required due to the valve working from LCP-43.

23. De-energize BAM-113A at SSD-EMCC-312B, compartment 2J, located in Fire Area RAB8B. Then, manually open BAM-113A in Fire Area RAB39 to establish the RCS charging flow path from BAMT. (1 hour)

This action to operate BAM-113A was completed within 53 minutes from LCP-43. The contingency to operate the valve locally manually was not required due to the valve working from LCP-43.

24. De-energize BAM-113B at SSD-EMCC-311B, compartment 7C, located in Fire Area RAB8B. Then, manually open BAM-113B in Fire Area RAB39 to establish the RCS charging flow path from BAMT. (1 hour)

This action to operate BAM-113B was completed within 53 minutes from LCP-43. The contingency to operate the valve locally manually was not required due to the valve working from LCP-43.

25. Remove trip circuit control power fuse from EBKR-31B-5C at SSD-ESWGR-31B in Fire Area RAB8B, then manually operate the breaker as necessary to operate CVC-MPMP-0001B to support RCS inventory and pressure control while in hot standby. For cold shutdown, replace trip circuit control power fuse and operate pump from LCP-43. (1 hour)

This action to operate Charging Pump B was completed within 54 minutes from LCP-43. The contingency to operate the breaker for Charging Pump B locally manually was not required due to the pump working from LCP-43.

26. Monitor BAMU Tank levels using local indicators BAM-ILI-0240 and BAM-ILI-0241 in fire area RAB39. (1 hour)

This action was not performed as soon as Charging Pump B was started. There is no time critical basis in Appendix D for performing this function within 1 hour of evacuating the Control Room. This action would have been performed within 20 minutes of starting Charging Pump B, in accordance with step 45, which requires recording initial BAM Tank levels. Based on minimum Tech Spec volumes in the BAM Tanks, 20 minutes would be sufficient time to perform this function.

27. Restore Pressurizer Proportional Heater Bank 2 as follows: (1 hour)
a) Verify ALL breakers on SSD-ESWGR-32B are open.

b) Place SST 3B32 FDR PERMISSIVE TO CLOSE Key Switch, to SWGR

c) At SSDEBKR3B-9, Switchgear 32B Supply, depress AND hold PB/Perm pushbutton, THEN place breaker hand switch to Close

d) Verify SSDEBKR3B-9, Switchgear 32B Supply, indicates closed.

Heaters are operated (On/Off) from LCP-43.

This action was completed within 55 minutes. There is no time critical basis in Appendix D for performing this action. Pressurizer heaters are not required for safe shutdown of the plant. This step was placed at this point in the attachment for prioritization. This is due to the fact that it makes sense to place it in the procedure after starting Charging and after the 'B' train is energized.

Place fuse selector switch SS-1 To BACKUP for CVC-216B at ID-EPDP-91B in Fire Area RAB7B and reset CVCEBKR91B-31 if necessary. This will restore operation of the valve to LCP-43. (1 hour)

This action was not performed since it was not necessary for the given scenario. CVC-216B was energized on LCP-43.

28. At CHWEBKR311B-5M on SSD-MCC-311B in Fire Area RAB8B, select the backup control power fuse for Essential Chilled Water Pump B, CHW-MPMP-0001B, and start locally in RAB2. (90 minutes)

This action was not performed since it was not necessary for the given scenario. This step to start Essential Chiller B was completed within 44 minutes.

29. Open breaker 18 at LVD-EPDP-61B in Fire Area RAB8B to provide HVAC to the SWGR Rooms, Cable Vault, and Battery Rooms. This action will fail CHW-900 to the fully open position. It will also fail SVS-103B, SVS-105B, and SVS-106B in the required open position. (90 minutes)

This step was completed within 32 minutes.

30. Reset control power breaker at LVD-EPDP-61B, circuit no. 11 in Fire Area RAB8B. This will restore operation of CC-301B, CC-322B, ACC-112B, and ACC-139B to LCP-43 to provide essential mechanical support. (90 minutes)

This step was completed within 41 minutes.

31. To establish H&V Equipment Room HVAC, open doors to RAB2 and operate HVR-MFAN-0024B from its local control station in RAB2. (90 minutes)

This step was completed within 55 minutes.

32. To establish Battery Room ventilation, manual action will be required to open Battery Room Doors in Fire Area RAB8C. (90 minutes)

This step was completed within 60 minutes.

33. Operate SVS-MAHU-0001B from its local control station located in Fire Area RAB7B, and Alternative Shutdown is credited for this Fire Area. Selection of backup fuses at SSD-EMCC-313B, compartment 5H, located in Fire Area RAB8B, is required. (90 minutes)

This step was completed within 66 minutes.

34. A spurious EFAS1-A-AUTO signal is addressed as follows:

Feed to SG-1 will be manually controlled from LCP-43 using EFW-223A and EFW-224A. De-energization of the Train A buses will secure EFW-MPMP-0001A, and the EFW-MPMP-0001A/B turbine may be tripped locally in Fire Area RAB39 if/when necessary to mitigate the spurious opening of MS-401A.

Feeding of the Steam Generators from Emergency Feed Water was established and monitored within the first 15 minutes after evacuating the Control Room with EFW Pump B and EFW Pump AB.

35. A spurious EFAS1-B-AUTO signal is addressed as follows:
Feed to SG-1 will be manually controlled from LCP-43 using EFW-223A and EFW-224A. The EFW-MPMP-0001A/B turbine may be tripped locally in Fire Area RAB39 if/when necessary to mitigate the spurious opening of MS-401B.

Feeding of the Steam Generators from Emergency Feed Water was established and monitored within the first 15 minutes after evacuating the Control Room with EFW Pump B and EFW Pump AB.

36. A spurious EFAS2-A-AUTO signal is addressed as follows:
Feed to SG-2 will be manually controlled from LCP-43 using EFW-223B and EFW-224B. De-energization of the Train A buses will secure EFW-MPMP-0001A, and the EFW-MPMP-0001A/B turbine may be tripped locally in Fire Area RAB39 if/when necessary to mitigate the spurious opening of MS-401A.

Feeding of the Steam Generators from Emergency Feed Water was established and monitored within the first 15 minutes after evacuating the Control Room with EFW Pump B and EFW Pump AB.

A spurious EFAS2-B-AUTO signal is addressed as follows:

Feed to SG-2 will be manually controlled from LCP-43 using EFW-223B and EFW-224B. The EFW-MPMP-0001A/B turbine may be tripped locally in Fire Area RAB39 if/when necessary to mitigate the spurious opening of MS-401B.

Feeding of the Steam Generators from Emergency Feed Water was established and monitored within the first 15 minutes after evacuating the Control Room with EFW Pump B and EFW Pump AB.

37. To mitigate spurious SI-MPMP-0002B and CS-MPMP-0001B actuation, remove control power from breakers EBKR-3B-3 and EBKR-3B-5 and manually open the breakers at 4KV-ESWGR-3B in Fire Area RAB8B.

This action was completed within 30 minutes.

38. Connect Excore Neutron Flux Channel D to provide indication in RAB9. Utilize the existing disconnect fittings at the preamplifier in Fire Area RAB5 and perform steps as specified in the Control Room Evacuation with Fire Procedure.

This action was completed within 74 minutes.

ATTACHMENT 2

8-26-2003 Appendix R Drill

This document discusses the performance and time results obtained during a drill on Revision 8 Change 0 of OP-901-502, Control Room Evacuation Procedure, which was requested by the NRC Inspection Team. The times for procedure step completion were recorded by personnel from the Waterford 3 Quality Assurance (QA) Department. Additional information, based on various critiques, is provided to assist Waterford 3 in improving OP-901-502 and drill related performance for future drills or inspections related to Appendix R compliance.

APPENDIX R DRILL HELD ON 8/26/2003

A drill was conducted at Waterford III SES at the request of the NRC to test the site's response to a fire in the Control Room [Fire Area RAB1 (Includes RAB1A & 1E)] using OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown procedure. The drill was performed with Operation's Shift 'C' providing the drill participants. The participants and drill positions are listed below:

Name	Department	Role
Brian Lietzke	Operations	Shift Manager
Ernie Viator	Operations	Control Room Supervisor
John Jarrell	Operations	Primary NPO
Tracey Berry	Operations	Secondary NPO
Danielle Dale	Operations	Shift Technical Advisor
Pete Bocskov	Operations	NAO/Emergency Communicator

The Drill Monitors and Observers consisted of the following personnel:

Name	Department / Organization	Role
Robert Simpson	Operations	Lead Monitor – LCP43
Paul Wood	Operations	Field Monitor
Ciro J. Dimarco	QA	Timer – Secondary NPO
Phil Kelly	QA	Timer – CRS
Mike Mason	QA	Timer – LCP43
John Burke	QA	Timer – NAO
Barbara Morrison	QA	Timer – LCP43
John Mateychick	NRC	Lead Observer – LCP43
Geoffrey Miller	NRC	Observer – CRS
Ray Mullikin	NRC	Observer – Secondary NPO
Joe Taylor	NRC	Observer – LCP43

DRILL CONTROLS

The scenario was provided and known prior to the drill only by the NRC. The Operation's Field Monitor requested the NRC Inspection Team provide the scenario to drill monitors prior to the drill so that proper cues and indications could be given to the participating shift crew. That request was denied.

The Operations Lead Monitor provided the cues and parameters to the shift crew at the remote shutdown panel (LCP43) during the drill. The NRC Lead Observer provided cues to the Lead Monitor at LCP43. The Operations Field Monitor provided cues to operators when no NRC monitor was present to provide them.

Prior to the drill, the NRC Inspection Team was informed that delays would be encountered because of the following issues:

- Non-standard methods would be utilized for communications.
- Routing requirements would be different than during an actual Appendix R event.
- RAB Ventilation would be removed from service for maintenance, providing a great potential for delays in exiting the CAA (Controlled Access Area).
- Operators would need to answer questions or explain actions to the observers as requested.

All of these conditions would result in delays that would not occur in an actual event. The NRC Inspection Team acknowledged that these delays would be considered when evaluating procedure performance against requirements.

A brief was held in which the NRC Inspection Team, operation's drill participants and monitors, QA personnel, and Operation's Management attended. The above listed delays were discussed along with the current actual plant configuration. The need to focus on safety was stressed during the brief.

The times for the actions performed in this drill were compiled by the QA Department monitors. The times in the documentation provided to the NRC were raw times (times that included delays due to the logistic issues previously mentioned).

REQUIREMENTS

Calculation EC-F00-026 provides the Post Fire Safe Shutdown Analysis which demonstrates compliance to 10CFR50 Appendix R. The purposes of Appendixes C and D from this calculation are summarized below:

- Appendix C: Provides a fire area by fire area compliance summary, which includes credited equipment for the area, cable by cable analysis, and necessary actions to achieve post fire safe shutdown.
- Appendix D: Provides the Functional Requirements Time Line Analysis that prioritizes the actions required to prevent the plant from being placed in an unrecoverable condition.

Appendix D supports that only actions necessary to be performed are required to meet the respective time requirements. As an example, deenergizing the 7kV Buses is required within 10 minutes of a Normal Spray Valve failure. The time requirement does not exist unless the failure actually occurred. Operators are trained to recognize component failures utilizing system response indications. Emergency Operating Procedures, which all licensed operators receive simulator training on throughout the year, are safety function based, such that operators prioritize actions based upon safety function importance.

Therefore recognizing a Normal Spray Valve, Atmospheric Dump Valve, or Main Steam Isolation Valve failure (as examples) is engrained in operators via the training process. Additionally, parameters have been added to OP-901-502 to assist in recognizing the respective component failures.

GENERAL DISCUSSION OF DRILL PERFORMANCE AND DELAYS ENCOUNTERED

At the beginning of the drill, the Lead Observer informed the Lead Monitor that the #1 Atmospheric Dump Valve (ADV) would fail open when the crew arrived at LCP43. After the crew established communications with the +35 Relay Room, the PNPO started reviewing system parameters on LCP43. He noticed #1 ADV was fully open (he noted that steam pressure was lower on the #1 SG than #2 SG and was continuing to go lower) and that he could not take manual control of the #1 ADV.

At this point the PNPO called the Secondary Nuclear Plant Operator (SNPO) in the +35 Relay Room to dispatch him to the #1 ADV to obtain manual control of the valve. The time recorded by QA was nine minutes. The actual time considering delays (that would not be encountered in an actual event) was less than 6 minutes. The delays included a two minute delay while the crew at LCP43 waited for the +35 Relay Room personnel to call them on the phone, and vice-versa. In an actual event the two groups would have been on sound powered phones and no delay would have existed (sound powered phones were not used for the drill because actuation of the sound powered phone system would have impacted normal Control Room communication.)

A second delay of two minutes was encountered when the PNPO had to find the phone number and call the SNPO to dispatch him to the #1 ADV. The SNPO had to stop transferring controls from the Control Room to LCP43, answer a phone, talk to the PNPO and then go back and turnover the information to the Control Room Supervisor (CRS). In an actual event, use of the sound powered phones would have allowed essentially instantaneous communication.

The CRS then completed transferring the Control Room to LCP43. This consisted of 7 FR (transfer) switches in 3 cabinets and 3 key operated isolation switches on LCP80. The CRS then stopped performing the procedure to explain the FR switch alignment to the observers who had expressed confusion due to multiple switches with similar names. The CRS then waited for the PNPO to call him in the +35 Relay Room. In a real event, the CRS would have been on sound powered phones and the PNPO could have given him the instruction to perform his next step which was also in the +35 Relay Room. There was a 10 minute delay between the CRS completing step 3.3 and starting step 9. In actuality, the delay would have been less than one minute due to the close proximity of these actions in the +35 Relay Room. Our review deducted an additional 7 minutes from the QA time to account for this delay.

The Nuclear Auxiliary Operator (NAO) arrived in the TGB Switchgear Room and established communication with LCP43 within 5 minutes of evacuating the Control Room. The NAO's responsibility was to deenergize the 7kV busses that energized the Reactor Coolant Pumps in order to preclude a spurious spray actuation of normal pressurizer spray flow. The PNPO knew that an operator was available to deenergize the 7kV buses, and since there was no indication of a spray valve failure, the PNPO therefore focused on responding to the #1 ADV failure. Once the PNPO had established manual control of #1 ADV, the focus shifted to deenergizing the 7kV buses.

Had there been indications of a spray valve failing open, an operator was on station and was prepared to take the action necessary to secure spray flow by deenergizing the Reactor Coolant Pumps. Our analysis states "...actions to secure spurious spray flow should be completed within 10 minutes." Because there was no indication of spurious spray flow, the action was completed as required.

The next delay occurred when the CRS attempted to go to Emergency Diesel Generator B to perform steps to isolate its control from the Control Room. The QA timer allowed for a conservative 8 minute delay that accounted for problems logging into the CAA, and the additional time for walking from the RAB+21 to the RAB-4 and then back to the RAB+21. In an actual event the CRS would have walked through the Motor Generator Set Room door on the RAB+21, which is a direct route into the Controlled Access Area and which is procedurally directed.

The Operations Lead Monitor provided cues that the 3B safety bus was energized. There was no immediate feedback from the NRC Lead Observer that this was an erroneous cue for the intended scenario. At a later time the NRC Lead Observer informed the Operations Lead Monitor that when the CRS performed step 13 of OP-901-502, the Emergency Diesel Generator (EDG) B would not automatically start. At this point simulated plant conditions were such that this could only occur if EDG B tripped or failed to start automatically when the FR switches were actuated during the earlier step 3.2 (during the first 10 minutes of the drill). The failure would have been realized by the crew much earlier in the scenario if appropriate cues had been provided.

When EDG B fails to start automatically, step 14 of OP-901-502 is required to be performed as a contingency. Step 14 resets control power for 3B Switchgear UV relays, which would allow EDG B to be restarted manually. If it is assumed EDG B tripped due to spurious actuation, then this breaker was reset and EDG B manually restarted within the time requirements of the analysis.

During a later meeting, the NRC Inspection Team stated that the 3B bus was energized by offsite power and was the reason the 3B bus was still energized. The 3B bus was divorced from offsite power during step 3.2 of OP-901-502 when Isolation Switch FR-2 was actuated on Aux Panel 2B. This caused EDG B to start and load on the mini-sequencer. Again, with proper cues at the appropriate times, the failure could have been detected much earlier.

One observation made by the NRC Inspection Team is that Emergency Feedwater (EFW) flow had not been established within the time requirements of the analysis. This observation was based on the performance of step 16 of OP-901-502, which was performed at approximately 30 minutes into the drill. Step 16 places the EFW flow controllers in manual to feed the steam generators. However, the Operation's Lead Monitor observed that the flow was established within the time requirements. The PNPO verified the proper Steam Generator levels with adequate flow from EFW during the review of the control boards after the initial transfer of control to LCP-43, within the first 10 minutes. The PNPO verified EFW Pump A/B (steam drive pump) and EFW pump B (motor driven) were running and supplying both steam generators continuously throughout the scenario. The PNPO frequently requested cues from the monitor on what the values were for various parameters during this time. The EFW flow had been established and the PNPO did not see the need to take manual control of the valves since they were operating properly and isolated from the Control Room

It should be noted that the Compliance Assessment Summary of Appendix C, which the NRC was using as their timing guide, requires the FR switches to be thrown within 25 minutes. These switches were actuated during the first ten minutes of the drill.

Appendix C also required breaker EFWEBKRA 23 to be opened to verify EFW-229B fails open. The procedure used for the drill did not have this step in it. The PNPO verified that he had proper EFW flow to the generators. When questioned after the drill what action he would have taken if EFW-229B indicated closed on LCP43, he stated he would have sent an operator up to gag the valve open, which is in accordance with operation's management expectations. As an enhancement, Revision 8 Change 1 to OP-901-502 was incorporated to add verification EFWEBKRA 23 open. This change was implemented on 8/21/2003, prior to performance of this drill.

Concerning multiple spurious component failures, in reference to Generic Letter 86-10 Question and Answer section 5.3.10, and the statements by Ken Sullivan in 1997 on this issue, only "one worst case spurious actuation or signal resulting from the fire" can occur on systems that are isolated from the fire.

CONCLUSION

Waterford 3 has added enhancements to OP-901-502 under Revision 8 Change 3, which was brought up by the NRC Inspection Team as well as Shift C during their post evolution brief. In particular, notes have been added to alert the operator of time critical steps that are less than one hour, and the situations that would make them critical. In addition, an attachment has been added to OP-901-502, to provide operators with a quick reference for time critical steps that are required within 30 minutes or less.

The differences in observations between the NRC observers and Waterford 3, is due to inadequacies in drill simulation. Taking into account delays that were discussed and incorrect indications provided to the shift, which caused some confusion, all actions were completed within time requirements of EC-F00-026, Post Fire Safe Shutdown Analysis.

Waterford 3 is confident OP-901-502 can be successfully performed by operation's personnel to place the plant in a safe condition and meet all safety analysis requirements.

ATTACHMENT 3

Validation Walkdown
Of
OP-901-502

Evacuation of Control Room and Subsequent Plant Shutdown

OP-901-502 Time Validation

The operators performing the steps in the OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, will be at a minimum the members of the on-shift Control Room staff (SM, CRS, PNPO, SNPO and STA) and one Nuclear Auxiliary Operator who is filling the Emergency Communicator position. The five Nuclear Auxiliary Operators who make-up the on-site Fire Brigade were not considered when evaluating the time it would take to perform OP-901-502. More operators may be available to perform this procedure based on shift manning at the time, however, for the purposes of this time validation on-shift Control Room staff and the Emergency Communicator are the only ones being considered during this review.

Steps in OP-901-502 are sequenced according to the Compliance Assessment Summary of the calculation EC-F00-026, Post Fire Safe Shutdown Analysis. The steps that were evaluated per this time validation of OP-901-502 were those required to begin feeding the Steam Generators with Emergency Feedwater. These steps do not include the Immediate Operator Actions performed in the Control Room prior to the evacuation. The timing began at the step in the procedure when the Control Room staff would exit the Control Room.

Some of the steps in this procedure can be performed concurrently as directed by the SM/CRS. The steps that were considered to be performed concurrently are displayed on the OP-901-502 Time Sheet Validation chart.

The step number (Step No.) listed on the OP-901-502 Time Validation Worksheet, is a cumulative list of the steps in the procedure starting from when the operators leave the Control Room. It is does not correspond to a specific step number listed in the procedure. The step number does correspond to the page number of the Local Action Validation Walkdown.

The time to perform individual listed steps (Time to Perform Step (minutes)) on the OP-901-502 Time Validation Worksheet, is a representative time for an operator to walk to the location, read the step and then perform the action. These times are considered conservative because it is a walking validation which does not represent the urgency when performed under Emergency Conditions.

The time to perform the steps in the procedure are displayed on the OP-901-502 Time Sheet Validation chart under the Cumulative Time with Concurrent Step Performance (minutes) heading. This is the time that it is expected to complete the steps necessary to feed the Steam Generators with EFW. It should be noted that it is expected that the AB EFW Pump, the DC powered EFW Control valves and the EFW Isolation valves are expected to be available after the completion of step 3. Therefore, EFW will be available to feed the Steam Generators if the SM/CRS deems the RCS Heat Removal safety function is in jeopardy due to the Steam Generators emptying.

The position listed in the "Position Performing Step" section of the OP-901-502 Time Validation Worksheet, is not necessarily the actual person performing this step. It is however, a representative sample of a position that is available to perform this step. The operators in the field will perform the steps as they are available or as they are directed by the Auxiliary Control Room.

The Comments section of the OP-901-502 Time Validation Worksheet is a section where editorial comments were added to provide clarifying information. In some instances it was indicated where steps were not likely to be performed because the steps were contingency step. Contingency steps are provided in the procedure to address potential spurious or maloperation. Although the capability exists to respond to several of these contingencies the requirement is to respond to any but only one at a time. It should also be noted that several steps are confirmatory steps. For example the Reactor Coolant pumps are tripped prior to leaving the Control Room. Confirmatory actions are taken in the TGB to secure the 7kV buses feeding the RCPs in the unlikely event cable damaged occurs prior to Control Room evacuation."

The conclusion from the time validation of OP-901-502 is that the designated Operations personnel can perform the steps of the procedure and begin to feed the Steam Generators within 25 minutes. This includes completing all steps to preserve the Safety Functions up to that point in the procedure with the minimum shift complement.

Prepared By:	/ Robert E Simpson Jr.	Date: 7-25-2003
Reviewed By:	/ Terry Schreckengast	Date: 7-28-2003

OP-901-502 TIme Validation Worksheet

Step No.	Time to Perform Step (minutes)	Cumulative Time (minutes)	Cumulative Time with Concurrent Step Performance (minutes)	Position Perfoming Step	Comments
1	2.00	2.00	2.00	PNPO	This step will be performed by the PNPO after the PNPO the SM and the STA have arrived at LCP-43. Upon arrival, the procedure will be removed from the locker and begun to be performed.
2	0.50	2.50	2.75	STA	The STA will perform the role of the Emergency Communicator duing this event.
3	0.25	2.75		PNPO and SM	
4	0.25	3.00		PNPO and SM	
5	0.25	3.25		SM	
6	0.25	3.50	3.00	PNPO	
7	8.00	11.50	11.00	CRS and SNPO	As soon as communication is established between LCP-43 and the +35 Relay Room, the CRS and SNPO will commence transferring control from the Main Control Room to the Aux. Control Room (LCP-43).
8	1.00	12.50		PNPO and SM	This step will be performed concurrently with the previous step. As soon as the the switches transfer control to the Aux. Control Room, the corresponding switch on LCP-43 will light-up. At this point the PNPO can match the switch position and take control of the equipment.
9	1.00	13.50		PNPO and SM	
10	1.00	14.50		PNPO	
11	0.00	14.50	11.00	CRS	This step is a contingency step that is not expected to have to be performed. If the step is necessary the CRS will walk down from the +35 Relay Room to the B Switchgear to perform this evolution.
12	4.50	19.00	15.50	NAO and SNPO	The NAO will be sent to the TGB Switchgear at the start of the event and will ready to deenergize the 1A and 1B bus when called upon. The time listed is for the the SNPO to walk down to the TGB Switchgear and assist the NAO as necessary.
13	2.00	21.00		CRS	The CRS is expected to perform this evolution concurrent to the step that deenergizes the 1A and 1B bus.
14	0.25	21.25		CRS	The CRS is expected to perform this evolution concurrent to the step that deenergizes the 1A and 1B bus.
15	0.25	21.50		CRS	The CRS is expected to perform this evolution concurrent to the step that deenergizes the 1A and 1B bus.
16	1.00	22.50		CRS	The CRS is expected to perform this evolution concurrent to the step that deenergizes the 1A and 1B bus.

OP-901-502 TIme Validation Worksheet

Step No.	Time to Perform Step (minutes)	Cumulative Time (minutes)	Cumulative Time with Concurrent Step Performance (minutes)	Position Perfoming Step	Comments
17	2.00	24.50	17.50	NAO and SNPO	This step will either be performed by the NAO and SNPO, or started by the CRS until the the SNPO and NAO arrive.
18	0.00	24.50		CRS	This step is a contingency step that is not expected to have to be performed. If the step is necessary the CRS will perform this evolution.
19	2.00	26.50		CRS	
20	5.00	31.50	22.50	PNPO and SM	The PNPO and SM will be directing this procedure form the Aux. Control Room and will be ready to perform this step as the equipment becomes available. The AB EFW Pump should be available from the beginning of this event. The EFW Isolation and EFW Control valves are DC powered, therefore they should be available from the beginning of this event. Control of EFW flow to the SGs can be established by the PNPO and SM as the SM determines there is a need to preserve the RCS HEat Removal safety function.
21	5.00	36.50		CRS and SNPO	This step will be performed by the SNPO and CRS concurrently with the step to establish EFW flow.

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E0 Subsequent Operator Actions: General)

Step Number 1.1 Step Description Transfer ALL SIX comm. System control transfer panel switches to Pos 2.

NOTE

Time required for local action is the time interval required for operator to travel from the Control Room to the location, simulate or perform the local action.

Time Required to perform the local action: 2 minutes

Performed by: / Robert E Simpson Jr. (Sign / Print) 7-22-2003 Date

Reviewed by: / T. Schreckengast (Sign / Print) 7-27-2003 Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E0 Subsequent Operator Actions: General)

Step Number 1.2 Step Description Make page.

NOTE

Time required for local action is the time interval required for the operator to read the step and simulate or perform the local action.

Time Required to perform the local action: 0.50 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-21-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E0 Subsequent Operator Actions: General)

Step Number 2 Step Description Advise Shift Manager to implement E-plan.

NOTE

Time required for local action is the time interval required for the operator to read the step and simulate or perform the local action.

Time Required to perform the local action: 0.25 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-22-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E0 Subsequent Operator Actions: General)

Step Number 3 Step Description Go to Subsection E2.

NOTE

Time required for local action is the time interval required for the operator to read the step and simulate or perform the local action.

Time Required to perform the local action: 0.25 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-21-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 1 Step Description As staff becomes available, verify completion of Attachment 9, Hot Standby Outside Operations.

NOTE

Time required for local action is the time interval required for the operator to read the step and simulate or perform the local action.

Time Required to perform the local action: 0.25 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-22-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 2 Step Description Verify All DCT Fan Control Switches in Fast at LCP-43.

NOTE

Time required for local action is the time interval required for the operator to read the step and simulate or perform the local action.

Time Required to perform the local action: 0.25 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-27-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 3 Step Description Transfer Control Room Panel controls from Control Room to LP-43.

NOTE

Time required for local action is the time interval required for the operator to read the step and simulate or perform the local action.

Time Required to perform the local action: 8 minutes

Performed by: / Robert E Simpson Jr. (Sign / Print) 7-22-2003 Date

Reviewed by: / T. Schreckengast (Sign / Print) 7-27-2003 Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 4 Step Description Push to activate controls at LCP-43.

NOTE

Time required for local action is the time interval required for the operator to read the step and simulate or perform the local action.

Time Required to perform the local action: 1 minute

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-29-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 5 Step Description Verify controls transferred properly.

NOTE

Time required for local action is the time interval required for the operator to read the step and simulate or perform the local action.

Time Required to perform the local action: 1 minute

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-27-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 6 Step Description Manually Control ADVs to maintain SG pressure at 1000 psia, from LCP-43.

NOTE

Time required for local action is the time interval required for the operator to read the step and simulate or perform the local action.

Time Required to perform the local action: 1 minute

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7/22/03
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 7 Step Description Reenergize MSIV steam dumps if necessary.

NOTE

Time required for local action is the time interval required for operator to travel from the +35 Relay Room to B Switchgear and simulate or perform the local action. (This step is a contingency action and should not have to be performed during the event).

Time Required to perform the local action: 0 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-27-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 8 Step Description Deenergize SWGR 1A and SWGR 1B.

NOTE

Time required for local action is the time interval required for operator to travel from the +35 Relay Room to TGB Switchgear and simulate or perform the local action.

Time Required to perform the local action: 4.5 minute

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-27-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 9 Step Description Open breakers listed on Attachment 14, Loads to be Stripped From PDP 91B.

NOTE

Time required for local action is the time interval required for operator to travel from TGB Switchgear to the +35 Relay Room and simulate or perform the local action.

Time Required to perform the local action: 2 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-22-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 10 Step Description Verify Closed CC EBKR91B 15.

NOTE

Time required for local action is the time interval required for the operator to read the step and simulate or perform the local action.

Time Required to perform the local action: 0.25 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-27-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 11 Step Description Reset CVCEBKR91B 29 if necessary.

NOTE

Time required for local action is the time interval required for the operator to read the step and simulate or perform the local action. (This step is a contingency action and should not have to be performed during the event).

Time Required to perform the local action: 0.25 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-27-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 12 Step Description Open breakers on SUPS AB

NOTE

Time required for local action is the time interval required for operator to travel from the +35 Relay Room to AB Switchgear and simulate or perform the local action.

Time Required to perform the local action: 1 minute

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-22-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 13 Step Description Locally start EDG B.

NOTE

Time required for local action is the time interval required for operator to travel from the AB Switchgear to EDG B Room and simulate or perform the local action.

Time Required to perform the local action: 2 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-21-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 14 Step Description Reset control power for 3B Swgr UV relays and locally start EDG B, if necessary.

NOTE

Time required for local action is the time interval required for operator to travel from the EDG B Room to B Switchgear and simulate or perform the local action. (This step is a contingency action and should not have to be performed during the event).

Time Required to perform the local action: 0 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-27-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 15 Step Description Verify proper electrical alignment of Swgr 31B, MCC 315B supply and power to EG B Room Exh Fan Damper.

NOTE

Time required for local action is the time interval required for operator to travel from the EDG B Room to B Switchgear and simulate or perform the local action.

Time Required to perform the local action: 2 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-27-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 16 Step Description Establish EFW flow to Both SGs.

NOTE

Time required for local action is the time interval required for the operator to read the step and simulate or perform the local action.

Time Required to perform the local action: 5 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-27-2003
(Sign / Print) Date

LOCAL ACTION VALIDATION WALKDOWN

Proc. Number and Title OP-901-502, Evacuation of Control Room and Subsequent Plant Shutdown, (Section E2 Control Room Evacuation with fire),

Step Number 17 Step Description Verify MSIV Upstream Drip Pot. Bypass Drain Valves (MS-119A and MS-119B) are Closed.

NOTE

Time required for local action is the time interval required for operator to travel from the B Switchgear to the MSIV Area and simulate or perform the local action.

Time Required to perform the local action: 5 minutes

Performed by: / Robert E Simpson Jr. 7-22-2003
(Sign / Print) Date

Reviewed by: / T. Schreckengast 7-27-2003
(Sign / Print) Date

ATTACHMENT 4

FSAR Section 9.5.1.4.3

9.5.1.4.2 Alternative Shutdown Capability

Alternative shutdown measures are provided for the Control Room and the Cable Vault area.

In the Control Room and Cable Vault areas, it is not physically possible to protect all redundant safe shutdown systems (cables) against the adverse effects of spurious signals, fires or fire suppression activities. In the event that a fire disables the Control Room or Cable Vault area, the auxiliary control panel (LCP-43) and other local control stations, which are in separate fire areas, provide the alternative shutdown capability. Alternative shutdown capability provides safe shutdown control functions and indications at the local panels that are electrically isolated or are otherwise separate and independent from the Control Room and Cable Vault area. Electrical isolation is ensured by the use of transfer and isolation switches.

Table 9.5.1-4 lists the minimum equipment required by performance goals, which is sufficient to achieve and maintain safe plant shutdown following a Cable Vault and Control Room fire.

9.5.1.4.3 Primary Coolant System Interfaces

Several low pressure systems are connected to the high pressure primary coolant system. In these instances, low pressure system isolation is provided by redundant, electrically operated valves. The design of these systems ensures that a fire induced LOCA cannot result from a single fire opening two valves in series at a high/low pressure interface.

The following low pressure systems are connected to the high pressure primary system and use redundant, electrically controlled devices to isolate or preclude rupture of any primary coolant boundary.

a) Letdown Isolation
b) Primary Sampling
c) Rector Coolant Gas Vent System
d) Shutdown Cooling Isolation

The letdown line is provided with four pneumatic valves in series, each of which is capable of isolating the RCS. They consist of the letdown stop valve, the letdown containment isolation valve inside containment, the letdown containment isolation valve outside containment, and one of two letdown control valves outside containment. All valves fail close on loss of air or loss of power. In the unlikely event of a fire induced spurious failure leading to a letdown control valve to fail open, the operator can isolate letdown using any one of the other three valves.

The RCS is provided with three independent sampling paths (Surge Line, RCS Hot Leg, and Pressurizer Steam Space). Each line has redundant pneumatic valves which fail close on loss of air or power. The valves located inside the containment are powered from the SB DC bus and the valves located outside containment are powered from the SA DC bus. In addition, each line has a restriction orifice designed to limit flow to less than the makeup capability of one charging pump in the event of any failure in the downstream line. Thus, these passive devices eliminate the possibility of a LOCA through these lines.

The Reactor Coolant Gas Vent System is provided with redundant solenoid valves that are normally key-locked closed in the Control Room and which fail closed on a loss of power. Both vent lines have restriction orifices which are designed to limit flow to less than the makeup capability of one charging pump in the event of any failure in the downstream lines. Thus, as stated for the sample lines, these passive devices eliminate the possibility of a LOCA through these vent paths.

The RCS is isolated from the Shutdown Cooling System (SDCS) (i.e., Residual Heat Removal System) by two valves, one motor operated and one hydrualic-pneumatic operated, in a series arrangement in each of the loop 1 and loop 2 shutdown cooling suction lines inside the containment. Motor operated valves 1SI-V1502B and 1SI-V15O4A are key-locked closed, fail-as-is valves. Hydraulic-pnematic valves ISI-V15O1A and 1SI-V1503A are key-locked closed, fail closed valves. These isolation valves are provided with

ATTACHMENT 5

Commitment Change Evaluation
for
Commitment Number A-24525

COMMITMENT CHANGE EVALUATION FORM

PART I

Commitment Number: A-24525 **Licensing Tracking Number:** 98-0006

Source Document: Parent 1: Letter W3F1-97-0229 and Parent 2: IR 97-15

Has the original commitment been implemented? ☐ **YES** [x] **NO, Notify Licensing**

Original Commitment Description:

Identify established Appendix R Emergency Lighted Pathways in Operations Safe Shutdown Procedure.

Summary of Justification for Change or Deletion:

This evaluation addresses two changes to the subject commitment.
1. Extension of docketed due date from 3/31/98 to 5/15/98, and
2. Limiting scope of inclusion of established paths in Ops Procedures to non-routine paths.

Justifications:
1. Due date extension: The Operations Department requested extending the due date for this commitment a month and a half. This extension is necessary due to task prioritization, lack of personnel in the operations procedure group and ongoing discussions with Design Engineering. In the meantime, as a compensatory measure in conjunction with this item, the Operators are carrying flashlights to light their path in the event of an Appendix R fire. Therefore the relatively brief extension will not significantly impact the operators ability see their way to safe shutdown equipment during a fire.

2. Scope limitation for incorporation of established paths in Ops procedures: Operations has evaluated Design Engineering input from calculation EC-F98-001 and will incorporate one non-routine path from LCP-43 to the Controlled Access Area (CAA) into OP-901-502. Other paths are used on a routine basis by operators and do not need to be added to OP-901-502. The wording of the original commitment can be interpreted to mean that all established paths would be included in Ops procedures. This is considered to be an impractical interpretation, since it can not be determined where the Operators starting point will be at the onset of a fire event. Rather than try to include all possible paths, Operations is limiting the commitment to include only the path that is non-routine, that the operator may not be readily aware of. This change more accurately reflects the original intent of the statement.

(See attached copy of approved partial closure CCVF from Operations dated 3/31/98)

note: Original Ops Signatures provided on this attached from MKB

If this is a commitment revision, enter the revised commitment description, otherwise N/A:

Identify non-routine established Appendix R Emergency Lighted Pathways in Operations Safe Shutdown Procedure. Also, the due date for completion of the action changes from 3/31/98 to 5/15/98.

Prepared By: Oscar P. Pipkins / Oscar P. Pipkins **Date:** 3/31/98
Print Name/Signature

Supervisor Concurrence: N/A **Date:** 3/31/98
Print Name/Signature

Licensing Supervisor Approval: M K Braden MKB **Date:** 3/31/98
Print Name/Signature

PART II

2.1 **Could the revised commitment negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?**

[x] **NO** **Briefly describe rationale. Do not answer the questions in PART III. Go to PART IV.**

Rationale:

In the event that an Appendix R fire were to occur prior to completion of corrective action, adequate compensatory actions are in place to light the Operators paths (flashlights).

☐ **YES** **Go to PART III.**

PART III

3.1 **Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?**

☐ **NO** **Briefly describe rationale. Go to Question 3.2.**

Rationale:

☐ **YES** **EXIT PROCESS. (The balance of this form is not to be completed.) Do not proceed with change, OR discuss change with NRC and obtain any necessary approvals.**

3.2 **Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?**

☐ **NO** **Briefly describe rationale. Go to Question 3.3.**

Rationale:

☐ **YES** **EXIT PROCESS. Do not proceed with change, OR discuss change with NRC and obtain any necessary approvals.**

3.3 **Does the revised commitment involve a significant reduction in a margin of safety?**

☐ **NO** **Briefly describe rationale. If questions 3.1, 3.2 and 3.3 are NO, then go to PART IV.**

Rationale:

☐ **YES** **EXIT PROCESS. Do not proceed with change, OR discuss change with NRC and obtain any necessary approvals.**

PART IV

4.1 Was the original commitment necessary for compliance with an Obligation?

☐ **NO** **Go to PART V. Do not answer questions 4.2 and 4.3.**

[x] **YES** **Go to question 4.2.**

4.2 Does the revised commitment preserve compliance?

☐ **NO** **EXIT PROCESS. Do not make change, OR apply for appropriate regulatory relief.**

[x] **YES** **Briefly describe rationale. Go to question 4.3.**

Rationale:

The adequate compensatory measure, coupled with the relatively short due date extension period and the fact that no change has been made to the intent to complete the action, supports preservation of compliance.

4.3 Has the original commitment been implemented?

[x] **NO** **Revise Commitment and EXIT PROCESS. Provide timely notification to Licensing. (Licensing will notify NRC.)**
[Note: Licensing notified NRC (Phill Harrell) on 3/31/98 via telephone]

☐ **YES** **Revise Commitment and EXIT PROCESS. (NRC will be notified of revised commitment in 10CFR50.59 Summary Report.)**

PART V

5.1 Was the original commitment (1) explicitly credited as the basis for a safety decision in a NRC Safety Evaluation Report, (2) made in response to a NRC Bulletin or Generic Letter, (3) made in response to a request for information under 10CFR50.54(f) or 10CFR2.204, or (4) identified as a long term corrective action in response to a NRC Notice of Violation?

☐ **NO** **Go to PART VI. Do not answer question 5.2.**

☐ **YES** **Go to question 5.2.**

5.2 Has the original commitment been implemented?

☐ **NO** **Revise commitment and EXIT PROCESS. Provide timely notification to Licensing. (Licensing will notify NRC.)**

☐ **YES** **Revise commitment and EXIT PROCESS. (NRC will be notified of revised commitment in 10CFR50.59 Summary Report.)**

PART VI

6.1 **Was the original commitment made to minimize recurrence of an adverse condition (e.g., a corrective action in a violation response or LER)?**

☐ **NO** **Revise commitment and EXIT PROCESS. (NRC will be notified of revised commitment in 10CFR50.59 Summary Report).**

☐ **YES** **Go to question 6.2.**

6.2 **Is the revised commitment necessary to minimize recurrence of the adverse condition?**

☐ **NO** **Briefly describe rationale. Revise commitment and EXIT PROCESS. (NRC will be notified of revised commitment in 10CFR50.59 Summary Report).**

Rationale:

☐ **YES** **Revise Commitment. Notify NRC of the revised commitment in the 10CFR50.59 Summary Report with one exception. If the commitment was made in response to a NRC violation less than two years ago, then direct Licensing to notify NRC of revised commitment.**

Commitment A 24525 **Unit** W3 **Source Doc** 97-0229.DOC 9/29/1997

	Type	Number	Section	Rev	Date
Parent 1	C	WF197-0229			9/29/1997
Parent 2	IR	97-15			8/28/1997

Summary IDENTIFY ESTABLISHED APPENDIX R EMERGENCY LIGHTED PATHWAYS IN OPERATIONS SAFE SHUTDOWN PROCEDURE

Keyword List

	Name	Department
Lead Person	HALL, L.	OPS
Support Person		
Licensing Lead	PIPKINS, O.	

Originated Date 10/1/1997
Sched. Complete 5/15/1998
Closure Date
Status OPEN

Priority 1 **NRC Submittal** N **Outage** **Agency** NRC-R4 **Mode**

Commitment details updated by ED LEMKE **on** 4/8/1998 9:54 am

Commitment related documents updated by ED LEMKE **on** 4/8/1998 9:57 am

Text

<<W3F1-97-0229>>
ESTABLISHED LIGHTED APPENDIX R PATHS WILL BE IDENTIFIED IN THE OPERATIONS SAFE SHUTDOWN PROCEDURE.

<<CCEF 98-0006>>

IDENTIFY NON-ROUTINE ESTABLISHED APPENDIX R EMERGENCY LIGHTED PATHWAYS IN OPERATIONS SAFE SHUTDOWN PROCEDURE.

Comments

AN EXTENSION WAS MADE ON THE DUE DATE FOR THIS ITEM VIA COMMITMENT CHANGE EVALUATION PROCESS. THE NRC (PHIL HARRELL) WAS NOTIFIED ON 3/31/98. (OPP 3/31/98)

Current Status

Commitment A-24525 (Related Documents)

Relationship	Doc Type	Doc Number	Doc Section/ Procedure Title	Doc Rev	Doc Date
SOURCE DOC	C	WF197-0229			9/29/1997
SOURCE DOC	IR	97-15			
REFERENCE	CCEF	98-0006			3/31/1998

COMMITMENT CHANGE NOTICE

TO: O.P. Pipkins

FROM: E.L. Lemke

DATE: April 7, 1998

Subject: Commitment Change Notice

Attached are copies of approved CCEF 98-0006 and revised commitment A-24525 for this CCEF. If you have any questions, please call Ed Lemke, extension 6349.

Ed Lemke

Entergy Operations	MANAGEMENT MANUAL VOLUME IV	SITE DIRECTIVE NO. W4.501 ATTACHMENT II REV. NO. 2 PAGE 1 OF 1

COMMITMENT CLOSURE VERIFICATION FORM

CMS I.D. A 24525 SOURCE DOCUMENT 97-0229.doc

EXTENT OF CLOSURE: FULL ☐ PARTIAL ☒

MEANS OF CLOSURE LIST THE CLOSURE DOCUMENTS (I.E., LETTER, PROCEDURE, ETC.) FROM WHICH OBJECTIVE EVIDENCE OF COMMITMENT CLOSURE CAN BE OBTAINED.

Operations Department requests extension of this commitment associated with changing operations procedure associated with Control Room evacuation to a new due date of 5/15/98. This extension is necessary due to task prioritization, lack of personnel in the operations procedure group and ongoing discussions with Design Engineering. Operations has evaluated DE input from calculation EC-F98-001 and will incorporate the non-routine path from LCP-43 to the CAA into OP-901-502. Other paths are used on a routine basis by operators and do not need to be added to OP-901-502. ☐ CONTINUATION SHEET

OBJECTIVE EVIDENCE FROM THE LISTED CLOSURE DOCUMENTATION MUST BE ATTACHED

Louis E Dall
COGNIZANT INDIVIDUAL

3/31/98
AUTHORIZED SIGNATURE/DATE
(SUPERVISOR/MANAGER)

LICENSING

THE ATTACHED OBJECTIVE EVIDENCE SUPPORTS THE EXTENT OF CLOSURE INDICATED ABOVE

Dean P. Watkins
LE SIGNATURE

3/31/98
DATE

Commitment A 24525 Unit W3 Source Doc 97-0229.DOC 9/29/1997

	Type	Number	Section	Rev	Date
Parent 1	C	WF197-0229			9/29/1997
Parent 2	IR	97-15			8/28/1997

Summary: IDENTIFY ESTABLISHED APPENDIX R EMERGENCY LIGHTED PATHWAYS IN OPERATIONS SAFE SHUTDOWN PROCEDURE

Keyword List:

	Name	Department
Lead Person	HALL, L.	OPS
Support Person		
Licensing Lead	PIPKINS, O.	

Originated Date: 10/1/1997
Sched. Complete: ~~3/31/1998~~ 5/15/98
Closure Date:
Status: OPEN

Priority: 1 NRC Submittal: N Outage: Agency: NRC-R4 Mode:

Commitment details updated by: OPIPKIN on: 10/1/1997 2:30 pm

Commitment related documents updated by: on:

Text

<<W3F1-97-0229>>
ESTABLISHED LIGHTED APPENDIX R PATHS WILL BE IDENTIFIED IN THE OPERATIONS SAFE SHUTDOWN PROCEDURE.

Comments

Current Status

ATTACHMENT 3

OP 901-502, Revision 8, Change 4

Timeline Validation

OP-901-502 Revision 8 Change 4 Timeline Validation

Timeline validation for OP-901-502 performance for evacuation of control room due to a fire. Timeline includes time to perform immediate operator actions. Actual timing for immediate operator actions was approximately 1 minute. 2 minutes was used for conservatism. As much as possible throughout the timeline, one position (CRS) was utilized to perform manual actions. This is conservative since this ensures at least two other individuals are available for support. Additionally the Shift Manager was utilized solely for the command and control function at LCP-43.

Timing was performed by Quality Assurance personnel as performed by a licensed operator. The actions and times were independently verified by a separate licensed operator. A Fire Protection Engineer performed a review of results against safe shutdown analysis requirements.

Performed by: _____ 12-8-03
(Licensed Operator Signature) (Date)

Timed by: _____ 12/8/03
(Quality Assurance Signature) (Date)

Verified by: _____ 12-8-03
(Licensed Operator Signature) (Date)

Reviewed by: _____ 12-8-03
(Fire Protection Engineering Signature) (Date)

SECTION AND STEP	TIME FOR STEP COMPLETION	DURATION (MINUTES)	NOTE	AFFECTED POSITION
D / 1-4	2.00	2.00		PNPO, SNPO, CRS, SM
E0 / 1-3	2.00	4.00		PNPO, SNPO, CRS, SM
E2 / 1-3	2.75	6.75	Step 3 completed within 10 minutes of evacuating Control Room.	PNPO, CRS
E2 / 4-6	2.50	8.50	Step 5 & 6 completed within 10 minutes of component failure.	PNPO, SNPO Note: Assumed either ADV #1 or MSIV #1 failure. SNPO dispatched at 6 minutes to ADV #1.
E2 / 7	1.50	7.50	Step 7 completed within 10 minutes of component failure.	SNPO Note: Assumed either ADV #1 or MSIV #1 failure. SNPO dispatched at 6 minutes for response.
E2 / 8	1.50	9.00	Step 8 completed within 10 minutes of component failure.	NAO Note: NAO arrived at TGB Switchgear at 7.50 minutes.
E2 / 9	1.75	8.50	Step 9 completed within 10 minutes of evacuating control room.	CRS (SNPO and NAO available, to be conservative – timed CRS only one available to perform actions)

SECTION AND STEP	TIME FOR STEP COMPLETION	DURATION (MINUTES)	NOTE	AFFECTED POSITION
E2 / 10-11	0.25	8.75		CRS (SNPO and NAO available, to be conservative – timed CRS only individual available to perform actions)
E2 / 12	1.25	10.00		CRS (SNPO and NAO available, to be conservative – timed CRS only one available to perform actions)
E2 / 13	2.75	12.75	Step 13-15 completed within 20 minutes of power loss.	CRS (SNPO and NAO available, to be conservative – timed CRS only one available to perform actions)
E2 / 14	3.00	15.75	Step 13-15 completed within 20 minutes of power loss.	CRS (SNPO and NAO available, to be conservative – timed CRS only one available to perform actions)
E2 / 15 - 15a	2.50	18.25	Step 13-15 completed within 20 minutes of power loss. Step 15a completed within 25 minutes of evacuating control room.	CRS (SNPO and NAO available, to be conservative – timed CRS only one available to perform actions)

SECTION AND STEP	TIME FOR STEP COMPLETION	DURATION (MINUTES)	NOTE	AFFECTED POSITION
E2 / 16	5.75	24.00	Step 16 completed within 25 minutes of evacuating control room.	CRS (SNPO and NAO available, to be conservative – timed CRS only one available to perform actions) Note: Assumed that EDG B failed to start, CRS restored EDG B to service, then EFW Pump B failed to start and EFW Pump AB tripped on overspeed.
E2 / 17	5.00	29.00	Step 17 should be completed within 30 minutes of evacuating Control Room.	CRS (NAO available, to be conservative – timed CRS only one available to perform actions after restoring EFW Pump AB for service).
E2 / 18 - 20	4.50	28.50	Step 19 should be completed within 30 minutes of evacuating the Control Room (for equipment protection only).	SNPO or NAO Note: Assumed that step performance did not commence until CRS had completed step 16 for restoring EFW Pump AB to service.

SECTION AND STEP	TIME FOR STEP COMPLETION	DURATION (MINUTES)	NOTE	AFFECTED POSITION
E2 / 21 - 23	18.5	47.50	Step 21 should be completed within one hour of evacuating the Control Room.	CRS (SNPO and NAO available, to be conservative – timed CRS only one available to perform actions)
E2 / 24	1.00	48.50		PNPO
E2 / 25	1.75	50.25	Step 25 completed within 60 minutes of evacuating control room.	CRS
E2 / 26	.25	50.50		CRS
E2 / 27-28	N/A	N/A		Steps 27 – 28 are deleted.
E2 / 29	8.00	58.50	Step 29 completed within 60 minutes of evacuating control room.	CRS
E2 / 30 -32	4.00	54.50	Step 30 – 32 to be completed within 60 minutes of evacuating control room.	SNPO Note: Assumed that step performance did not commence until CRS had completed step 26.
E2 / 33-34	2.00	N/A		PNPO Note: Step is continuously performed and has no affect on total duration.
E2 / 35-36	5.00	67.50		CRS
E2 / 37	3.00	70.50		CRS
38-39	1.00	N/A		PNPO Note: Step is continuously performed and has no affect on total duration.
40	5.00	75.50		PNPO and NAO