UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

November 09, 2007

Kevin T. Walsh
Vice President Operations
Waterford 3
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000382/2007004

Dear Mr. Walsh:

On October 7, 2007, the NRC completed an inspection at your Waterford Steam Electric Station, Unit 3. The enclosed report documents the inspection findings, which were discussed on October 4, 2007, with Mr. Joe Kowalewski and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents three findings of very low safety significance (Green). All of these findings were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these violations as noncited violations (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Waterford Steam Electric Station, Unit 3, facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response, if any, 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/

Jeff A. Clark, P. E.
Chief, Project Branch E
Division of Reactor Projects

Docket: 50-382
License: NPF-38

Enclosure: NRC Inspection Report 050000382/2007004
w/Attachment: Supplemental Information
Simplified Fire Risk Assessment for Hemyc Fire Wrap

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SUNSI Review Completed: _JAC____ ADAMS: √ Yes ☐ No Initials: _JAC____
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RIV:SRI:DRP/E	SPE:DRP/E	C:DRS/OB	C:DRS/EB2
DHOverland	GDReplogle	ATGody	LJSmith
E-JAC	***E-JAC***	***/RA/***	***/RA/ DProulx for***
10/30/07	10/30/07	11/01/07	10/30/07

C:DRS/PSB	C:/DRS/EB1	C:DRP/E		
MPShannon	WBJones	JAClark		
/RA/	***/RA/***	***/RA/***		
11/06/07	10/30/07	11/09/07		

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-382
License No.:	NPF-38
Report No.:	05000382/2007004
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	July 8 through October 7, 2007
Inspectors:	D. H. Overland, Acting Senior Resident Inspector G. Replogle, Senior Project Engineer G. L. Guerra, CHP, Health Physicist, Plant Support Branch G. Pick. Senior Reactor Inspector, Engineering Branch 2 P. J. Elkmann, Emergency Preparedness Inspector, Operations Branch
Approved By:	Jeff Clark, Chief, Project Branch E
ATTACHMENTS:	Supplemental Information

SUMMARY OF FINDINGS

IR05000382/2007-004; 07/08/2007 - 10/07/2007; Waterford Steam Electric Station, Unit 3;

The report covered a 3-month period of inspection by resident inspectors and a senior project engineer, a health physicist, a senior reactor inspector, and an emergency preparedness inspector. The inspectors identified three Green findings. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP 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.

Waterford 3 Plant formally committed to converting their Fire Protection Program to comply with the requirements of 10 CFR Part 50.48.(c) and National Fire Protection Association Standard 805. This involves using a risk-informed methodology. The conversion and licensing processes are expected to identify and address a variety of difficult issues that are normally the subject of triennial fire protection inspections. Since any findings in this area will be addressed under the new, rather than the existing, program, the NRC has adapted its inspection and enforcement of certain issues for plants in this situation. As a result, the scope of this inspection was modified and some issues raised in this inspection are documented but subject to enforcement discretion.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of Technical Specification 6.8.1.a (Procedures) for an inadequate boric acid evaluation procedure and for the failure to follow the same procedure. Specifically, the procedure noted that small amounts of boric acid could severely corrode carbon and low alloy carbon steel, but only had engineers check drawings for carbon steel components. Components with low alloy steel on the containment spray pumps were sometimes ignored. In addition, the procedure required pictures of the boric acid condition but, for some evaluations, no pictures were taken of the containment spray pump leaks. This made trending of the condition, to check for worsening, difficult. The inspectors determined that engineers were not following the boric acid evaluation procedure when performing the evaluations, they simply filled out the forms. The procedure contained valuable insights vital for proper boric acid evaluations, whereas the forms did not.

  The finding was more than minor because it could, if left uncorrected, result in a more significant safety concern. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheet, the finding was determined to have very low safety significance (Green) because it did not result in an actual loss of safety function for the containment spray system. The cause of the finding has a cross-cutting aspect in the area of human performance, work practices component, in that the licensee failed to

effectively communicate the expectations regarding procedural compliance and personnel follow procedures (H.4(b)) (Section 1R19).

- Green. The inspectors identified two examples of a noncited violation of Waterford Steam Electric Station, Unit 3 Facility Operating License Condition 2.C.9 for failure to implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility. In the first example, the pre-fire strategy for vital switchgear Room B did not contain adequate information regarding the doors required to be open to allow the desired ventilation flowpath, nor did it contain the required number of smoke ejectors necessary to desmoke the switchgear room in a manner that would allow the implementation of OP-901-524, "Fire In Areas Affecting Safe Shutdown." In the second example, the licensee did not take corrective actions for a previously identified issue in a timely fashion. Specifically, the deficiencies in the pre-fire strategy for vital switchgear Room B were first identified on August 21, 2006. The deficient procedure was not corrected until September 14, 2007, after the senior resident inspector discussed the non-conformance with licensee management. The licensee entered this deficiency into their corrective action program for resolution.

  The finding was more than minor because it was associated with the mitigating systems cornerstone objective (Protection Against External Factors) to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, Appendix F, Phase 1 initial qualitative screening, the issue screened as having very low safety significance because the compensatory manual action required to safely shut down the plant is not needed in order to reach hot shutdown. This finding had a crosscutting aspect in the area of problem identification and resolution. Specifically, the licensee's personnel corrective action process failed to take appropriate corrective actions to address the safety issue in a timely manner (P.1(d)) (Section 4OA2).

Cornerstone: Barrier Intergrity

- Green. The inspectors identified a noncited violation of Technical Specification (TS) 3.4.7 for multiple failures to complete a radiochemical analysis for EBAR (Average Disintegration Energy) determination within the required periodicity. Specifically, on thirteen out of fifteen occasions, the licensee had failed to complete the analysis and replace the old EBAR value with the new EBAR value within the TS required interval of 136 to 229 days. EBAR is the average of the sum of average beta and gamma energies per disintegration for isotopes, other than radioiodines, with half-lives greater than fifteen minutes. Daily RCS samples are compared to this calculated value in order to ensure that 10CFR50.67 dose limits at the site boundary are not exceeded in the event of an accident scenario. The licensee entered this issue into their corrective action program for resolution.

  The finding was more than minor because it was associated with the cladding performance attribute of the barrier integrity cornerstone and affected the cornerstone objective of providing reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheet, the finding was determined to have very

low safety significance (Green) because it only affected the fuel barrier. This finding had a crosscutting aspect in the area of human performance. Specifically, the licensee's personnel work practices failed to support human performance by ensuring that activity status and completion are properly documented (H.4(a)) (Section 1R22).

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status: The plant began the inspection period on July 8, 2007, at 100 percent power and remained at approximately 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown

a. Inspection Scope

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

- August 8, 2007: Essential chilled water system Train A
- August 27, 2007: Low-pressure safety injection system Train B
- September 5, 2007: Low-pressure safety injection system Train A

Documents reviewed by the inspectors included:

- OP-009-008, “Safety Injection System,” Revision 19
- OP-002-004, “Chilled Water System,” Revision 301

The inspectors completed three samples.

b. Findings

No Findings of significance were identified.

.2 Complete Walkdown (71111.04S)

a. Inspection Scope

The inspectors: (1) reviewed plant procedures, drawings, the Final Safety Analysis Report, Technical Specifications, and vendor manuals to determine the correct alignment of emergency diesel generator Train A; (2) reviewed outstanding design issues, operator work arounds, and open work requests to verify that outstanding issues

did not adversely affect the functionality of the system; and (3) verified that the licensee was identifying and resolving equipment problems in accordance with corrective action program requirements.

Documents reviewed by the inspectors included:

- OP-009-002, Revision 301, “Emergency Diesel Generator”
- TM-C629.0305, “Vendor Technical Manual for Cooper Bessemer Emergency Diesel Generator”

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Inspection

a. Inspection Scope

The inspectors walked down the six below listed plant areas to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the compensatory measures were commensurate with the significance of the deficiency; and (7) reviewed the Updated Final Safety Analysis Report to determine if the licensee identified and corrected fire protection problems.

- July 17, 2007: Fire Zones 8C, 11, 12, and 13
- July 19, 2007: Fire Zones RAB 15, 16, 17, 18, 19, 20, and 21
- July 23, 2007: Fire Zones RAB 33, 35, 36, 37, 38, and 39
- August 2, 2007: Fire Zones RAB 2, 23, 31, 32, and 39
- August 7, 2007: Fire Zones RAB 8B, 25, 39, Cooling Tower A, Cooling Tower B, and Fuel Handling Building

- September 12, 2007: Fire Zones RAB 1B, 8A

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

.1 Annual External Flooding

a. Inspection Scope

The inspectors: (1) reviewed the Updated Final Safety Analysis Report, the flooding analysis, and plant procedures to assess seasonal susceptibilities involving external flooding; (2) reviewed the Updated Final Safety Analysis Report and corrective action program to determine if the licensee identified and corrected flooding problems; (3) inspected underground bunkers/manholes to verify the adequacy of (a) sump pumps, (b) level alarm circuits, (c) cable splices subject to submergence, and (d) drainage for bunkers/manholes; (4) verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and (5) walked down the one below listed area to verify the adequacy of: (a) equipment seals located below the floodline, (b) floor and wall penetration seals, (c) watertight door seals, (d) common drain lines and sumps, (e) sump pumps, level alarms, and control circuits, and (f) temporary or removable flood barriers.

- September 20, 2007: Susceptibility of dry cooling tower components to external flooding

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

Training Observation

a. Inspection Scope

On August 21, 2007, the inspectors observed training of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The training scenario

involved several instrument failures and a loss of instrument air pressure, leading to a manual reactor trip in which two control element assemblies fail to insert. An ensuing loss-of-coolant accident requires a manual initiation of safety injection and containment spray.

Documents reviewed by the inspectors included:

- Simulator Scenario Number E-83, Revision 1
- Emergency Operating Procedure OP-902-000, Revision 10, “Standard Post Trip Actions”
- Emergency Operating Procedure OP-902-008, Revision 14, “Functional Recovery Procedure”
- Emergency Operating Procedure OP-902-002, Revision 11, “Loss of Coolant Accident Recovery”
- Emergency Operating Procedure OP-901-511, Revision 7, “Instrument Air Malfunction”

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- Safety Injection Tank leakage

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

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

.1 Risk Assessment and Management of Risk

a. Inspection Scope

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

- August 22, 2007: Planned maintenance outage of shield building ventilation Train A
- August 27, 2007: Planned maintenance outage of low-pressure safety injection Train A
- September 11, 2007: Planned maintenance outage of emergency feedwater Train B
- September 14, 2007: Planned surveillance activities for undervoltage and shunt trip coil testing for reactor trip circuit breakers

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plants status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the Updated Final Safety Analysis Report and design-basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any Technical Specifications; (5) used the Significance

Determination Process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

- July 20, 2007: Operability evaluation addressing Appendix R required emergency lighting
- July 25, 2007: Operability evaluation addressing a crack in the Fuel Handling Building ceiling
- July 27, 2007: Operability evaluation addressing pressurizer heater design
- August 1, 2007: Operability evaluation addressing dry cooling tower Train A sump pump low flow
- August 2, 2007: Operability evaluation addressing recurring primary side steam generator valve and loose parts monitor system alarms

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

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

- August 9, 2007: Corrective maintenance to replace Transducers 5 through 8 on main feedwater ultrasonic flow meter Number 2
- August 29, 2007: Corrective maintenance to replace a faulty relay in essential chiller Train A chilled water Pump 1 breaker

- September 18, 2007: Planned maintenance to stroke test containment atmospheric purge Valves 103 and 104 following breaker maintenance

- September 19, 2007: Planned maintenance to stroke test emergency feedwater Valve 228A following breaker maintenance

- September 20, 2007: Planned maintenance on emergency feedwater Pump A to change the oil and lube the pump

- September 12, 2007: Planned maintenance to clean boric acid from containment spray Pump B

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

Introduction. The inspectors identified a Green noncited violation of Technical Specification 6.8.1.a (Procedures) for an inadequate boric acid evaluation procedure and for the failure to follow the same procedure. Specifically, the procedure noted that small amounts of boric acid could severely corrode carbon and low alloy carbon steel, but only had engineers check drawings for carbon steel components. Components with low alloy steel on the containment spray pumps were sometimes ignored. In addition, the procedure required pictures of the boric acid condition but, for some evaluations, no pictures were taken of the containment spray pump leaks. This made trending of the condition, to check for worsening, difficult. The inspectors determined that engineers were not reading the boric acid evaluation procedure when performing the evaluations, they simply filled out the forms. The procedure contained valuable insights vital for proper boric acid evaluations, whereas the forms did not.

Description. During a plant tour on September 12, 2007, the inspectors observed boric acid deposits in the Train A and B containment spray pump shaft cavities. The deposits originated from the mechanical shaft seal area and collected on various components, including low alloy steel bolts on the mechanical seal assembly. The inspectors asked the licensee for the boric acid evaluations which were required by Procedure EN-DC-319, "Inspection and Evaluation of Boric Acid Leaks," Revision 0. The latest evaluation for Pump B was dated May 25, 2007 and the most recent evaluation for Pump A was dated March 10, 2006.

> NOTE: The licensee currently performs boric acid evaluations in accordance with Revision 1 of Procedure EN-DC-319. In addition, this report discusses historical boric acid evaluations performed in accordance with other versions of the procedure. With respect to the procedural content related to this violation, there were no meaningful differences between the procedures.

Inadequate Procedure: The inspectors identified that Procedure EN-DC-319 was inadequate, in that it did not require actions to address boric acid wastage on low alloy

steels. The forms used for the evaluation, "Attachment 9.3, Identification of Boric Acid Leakage," and "Attachment 9.4, Evaluation/Screening of Boric Acid Leakage," did not require the engineers to identify or evaluate the impact of boric acid on low alloy steels. The forms only required actions to address carbon steel components. Engineers that performed the evaluations stipulated that there was a difference between carbon steel and low alloy steel and they were not required by the procedure (forms) to evaluate the latter.

Low alloy steel is carbon steel with low amounts of selected alloys added to enhance material hardening characteristics. The small amount of alloys do not enhance resistance to boric acid wastage. For example, NRC Information Notice 80-27, "Degradation of Reactor Coolant Pump Studs," details an instance where another utility identified significant boric acid wastage of reactor coolant pump low allow steel closure studs.

Further, Procedure EN-DC-319 contained the following precaution:

> Small amounts of boric acid have the potential to severely corrode high temperature carbon and low allow steel over a long period of time.

The inspectors had noted that boric acid was in contact with low alloy steel bolts on both containment spray pump mechanical seal housings but the boric acid evaluations did not address this condition. The pattern of dry boric acid suggested that the leaks were traversing past and onto the bolt shanks and threads, which were not visible unless removed from the assembly.

The licensee did have some pictures for containment spray Pump B (but not for A) and the inspectors noted that the current Pump B boric acid pattern was consistent with pictures dated May 25, 2007. Since engineers did not consistently take pictures and documented leak descriptions were lacking detail, the impact and duration of the leaks was difficult to determine. At the present, the inspectors noticed some, but very limited, evidence of material wastage. Some bolt heads showed small amounts of external corrosion, but dried leakage past the bolt internals did not appear discolored. Therefore, currently pump operability was not in question.

The broader concern was that the same boric acid evaluation forms were used for all boric acid leaks and some components were more vulnerable to faulty evaluations than others. For example, components in containment are not readily inspectible and hot boric acid leaks on low alloy steel components could result in much more significant, but unaddressed, wastage. Therefore, the inspectors determined that this violation would be more significant if left uncorrected.

Pictures: The inspectors identified that the boric acid evaluator for the Train A, March 10, 2006 evaluation had failed to follow Procedure UNT-006-031, Revision 0 (a previous version of EN-DC-319), in that no pictures were taken to describe the condition. Procedure UNT-006-031 specified, in part:

If possible, include pictures of the overall component, close-up leakage conditions or any other relevant condition to assist in describing the condition and component location.

Contrary to the above, it was possible to include pictures of the overall component and boric acid build up but no pictures were taken. In addition, the description on the licensee's boric acid evaluation form was vague and it was impossible to tell if the leak had gotten worse or if the buildup of boric acid was stagnant for a sustained period.

Overall Assessment: While the licensee's procedure provided valuable information regarding the vulnerability of low alloy steel to boric acid wastage and the need to document boric acid leaks with pictures, engineers did not routinely follow the procedure. Instead, they accomplished their evaluations by simply filling out the form. The failure to properly use and implement the procedure was a significant contributor to the violation and its significance (more than minor).

Analysis. The failure to establish an adequate procedure for boric acid evaluations, and the failure to implement the procedure, were performance deficiencies. The finding was more than minor because it could, if left uncorrected, result in a more significant safety concern. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheet, the finding was determined to have very low safety significance (Green) because it did not result in an actual loss of safety function for the containment spray system. The cause of the finding has a cross-cutting aspect in the area of human performance, resources component, in that the licensee failed to effectively communicate the expectations regarding procedural compliance and personnel follow procedures (H.4(c)).

Enforcement. Technical Specification 6.8.1.a (Procedures) requires the licensee to establish and implement procedures recommended by Appendix A to Regulatory Guide 1.33, Revision 2, 1978. Appendix A, Section 9 recommends procedures for maintenance, including inspection. Procedure UNt-006-031 required, in part, pictures of boric acid leaks, if possible. Contrary to the above, it was possible to take pictures of the boric acid leak evaluated by the March 10, 2006 but no pictures were taken. In addition, Procedure EN-DC-319 was inadequate, in that it did not require engineers to evaluate the impact of boric acid on low alloy steels. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program (CR-WF3-2007-03590), it is considered a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000382/2007004-01, Inadequate Boric Acid Leak Evaluations.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report, procedure requirements, and Technical Specifications to ensure that the five below listed surveillance activities demonstrated that the structures, systems, and components tested were capable of performing their intended safety functions. The inspectors either

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

- July 26, 2007: Maintenance Procedure MM-007-010, Revision 15, Change 3, *Fire Extinguisher Inspection and Replacement*, is used to ensure that all site fire extinguishers are in working condition.

- August 7, 2007: Surveillance Procedure OP-903-003, Revision 11, Change 1, *Charging Pump Operability Check*, is used to ensure that charging Pump AB discharge pressure, flow, and vibration characteristics are within design parameters.

- August 7, 2007: Surveillance Procedure OP-903-035, Revision 12, *Containment Spray Pump Operability Check*, is used to ensure that containment spray Pump B discharge pressure, flow, and vibration characteristics are within design parameters.

- August 20, 2007: Surveillance Procedure OP-903-046, Revision 301, *Emergency Feedwater Pump Operability Check*, is used to ensure that emergency feedwater Pump AB discharge pressure, flow, and vibration characteristics are within design parameters.

- September 5, 2007: Chemistry Procedure CE-003-306, Revision 9, *Determination of the Average Beta-Gamma Energy of Reactor Coolant*, calculates the activity in the reactor coolant due to radioisotopes with a half-life of greater than 15 minutes.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed five samples.

b. Findings

Introduction. The inspectors identified a Green noncited violation of Technical Specification Surveillance Requirement 4.4.7 for multiple failures to complete a radiochemical analysis for EBAR (Average Disintegration Energy) determination within the required periodicity.

Description. Technical Specification Surveillance Requirement 4.4.7 requires:

> The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

Table 4.4-4, *Primary Coolant Specific Activity Sample and Analysis Program* item 3, requires a radiochemical analysis for EBAR determination to be completed once every 6 months.

EBAR is the average (weighted in proportion to concentration of each radionuclide in reactor coolant at time of sampling) of the sum of average beta and gamma energies per disintegration (in MeV) for isotopes, other than radioiodines, with half-lives greater than 15 minutes, making up at least 95 percent of total noniodine activity in reactor coolant. The EBAR value is then divided into a correction factor and utilized in a calculation to generate a value that establishes a maximum reactor coolant system (RCS) activity limit in microcuries per milliliter. Daily RCS samples are compared to this calculated value in order to ensure that 10 CFR 50.67 dose limits at the site boundary are not exceeded during an accident scenario.

On February 26, 2007, the RCS was sampled for EBAR relevant isotopes. Per procedure, several strontium and iron isotope samples were sent offsite for analysis. Results were complied and on May 1, 2007, the EBAR calculation was performed as a training performance evaluation for qualification of a chemistry technician. On September 5, 2007, the licensee noticed that the EBAR calculation had never been reviewed and the calculated value was never implemented for daily comparison. Condition Report CR-WF3-2007-3146 was generated.

In response to further questioning by the senior resident inspector about extent of condition, the licensee discovered that although the EBAR reactor coolant samples dating back to December 1999 were drawn on time, on thirteen out of fifteen occasions, the licensee had failed to complete the analysis and replace the old EBAR value with the new EBAR value within the Technical Specification-required interval of 136 to 229 days. On the two occasions that the Technical Specification requirement was met, it was only met due to the allowance of the 25 percent grace period. The average time for an EBAR value to be in place was 284 days, with the longest time period lasting 566 days.

The ability to sample, but fail to complete the analysis on time was due, in part, to the tracking method in place. A task to collect the EBAR sample is generated during the required periodicity. However, once the sample is obtained, the task is marked as complete and there are no additional tasks to ensure that the analysis of the sample or results calculation are completed.

Analysis. The failure to follow plant technical specifications and properly sample and analyze reactor coolant system chemistry to calculate a current EBAR value was a performance deficiency. The finding was determined to be NRC identified because it involved a previously documented licensee finding to which the inspector significantly added value. The finding was more than minor because it was associated with the cladding performance attribute of the barrier integrity cornerstone and affected the cornerstone objective of providing reasonable assurance that physical design barriers

(fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheet, the finding was determined to have very low safety significance (Green) because it only affected the fuel barrier. This finding had a crosscutting aspect in the area of human performance. Specifically, the licensee's personnel work practices failed to support human performance by ensuring that activity status and completion are properly documented (H.4(a)).

Enforcement: Surveillance Requirement 4.4.7 of Technical Specification 3.4.7 requires a radiochemical analysis for EBAR determination to be completed once every 6 months. Contrary to the above, on thirteen different occasions between January 2000 and September 2007, radiochemical analyses for EBAR determination were not properly conducted. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program (CR-WF3-2007-3301), it is considered a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000382/2007004-02, Missed Reactor Coolant System Chemistry Samples.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector performed an in-office review of Revision 22 to the Waterford 3 Emergency Plan Implementing Procedure EP-001-001, "Recognition and Classification of Emergency Conditions," Revision 22, received September 7, 2007. This revision added information about the choice of meteorological instruments used to measure the wind speed to the basis for Emergency Action Level HA6.

The revision was compared to its previous revision, to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, to Nuclear Energy Institute report 99-01, "Methodology for Development of Emergency Action Levels," Revision 4, to the requirements of 10 CFR 50.47(b), and to 50.54(q) to determine if the licensee adequately implemented 10 CFR 50.54(q). This review was not documented in a safety evaluation report and did not constitute approval of licensee changes, therefore the changes are subject to future inspection.

The inspector completed one sample during this inspection.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

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

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone
- Controls (surveys, posting, and barricades) of radiation, high radiation, and airborne radioactivity areas
- Radiation work permits, procedures, engineering controls, and air sampler locations
- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms.
- Barrier integrity and performance of engineering controls in airborne radioactivity areas
- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem Committed Effective Dose Equivalent
- Physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools.
- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection
- Corrective action documents related to access controls
- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies
- Radiation work permit briefings and worker instructions
- Adequacy of radiological controls such as, required surveys, radiation protection job coverage, and contamination controls during job performance
- Dosimetry placement in high radiation work areas with significant dose rate gradients

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

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

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

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

The inspector completed 21 of the required 21 samples.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

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

- Current 3-year rolling average collective exposure

- Work activities of exposure significance completed during the last outage

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

- Integration of ALARA requirements into work procedure and radiation work permit documents

- Shielding requests and dose/benefit analyses

- Dose rate reduction activities in work planning

- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding

- Source-term control strategy or justifications for not pursuing such exposure reduction initiatives

- Specific sources identified by the licensee for exposure reduction actions and priorities established for these actions, and results achieved against since the last refueling cycle

- Declared pregnant workers during the current assessment period, monitoring controls, and the exposure results

The inspector completed 10 of the required 29 samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Cornerstone: Mitigating Systems

The inspectors sampled licensee submittals for the three mitigating system performance index indicators listed below for the period of January 2006 through September 2007. The definitions and guidance of Nuclear Energy Institute 99-02, “Regulatory Assessment Indicator Guideline,” Revision 4, were used to verify the licensee’s basis for reporting each data element in order to verify the accuracy of performance indicator data reported during the assessment period. The inspectors reviewed licensee event reports, out-of-service logs, operating logs, and the Maintenance Rule database as part of the assessment. Licensee performance indicator data were also reviewed against the requirements of Procedure EN-LI-114, “Performance Indicator Process,” Revision 2.

- Emergency AC Power
- Support Cooling Water Systems
- Safety System Functional Failures

Occupational Radiation Safety Cornerstone

The inspector reviewed licensee documents from October 1, 2006, through June 30, 2007. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee’s technical specifications), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in Nuclear Energy Institute (NEI) 99-02). Additional records reviewed included as low as reasonably achievable (ALARA) records and whole body counts of selected individual exposures. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspector toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. Performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

- Occupational Exposure Control Effectiveness

The inspector completed the required sample (1) in this cornerstone.

Public Radiation Safety Cornerstone

The inspector reviewed licensee documents from October 1, 2006, through June 30, 2007. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to the NRC. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. Performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

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

The inspector completed the required sample (1) in this cornerstone.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

.2 Selected Issue Followup Inspection

a. Inspection Scope

In addition to the routine review, the inspectors selected the three issues, listed below, for a more in-depth review. The inspectors considered the following during the review of the licensee's actions: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences;

(4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

- September 18, 2007: Feasibility of manual compensatory actions in vital switchgear Room AB during a fire in vital switchgear Room B

- Access Control to Radiologically Significant Areas (Section 2OS1)

- ALARA Planning and Controls (Section 2OS2)

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

Introduction. The inspectors identified two examples of a Green noncited violation of Waterford Steam Electric Station, Unit 3 Facility Operating License Condition 2.C.9 for failure to implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility.

Description. Updated Final Safety Analysis Report (UFSAR) Section 9.5.1.3.1 provides a comparison of the licensee's Fire Protection Program to criteria described in Branch Technical Position APCSB 9.5-1, Revision 0, Appendix A.

.1 UFSAR Section 9.5.1.3.1.B.1, "Administrative Procedures, Controls and Fire Brigade," describes Branch Technical Position APCSB 9.5-1, Revision 0, Appendix A requirement that:

> Administrative procedures consistent with the need for maintaining the performance of the fire protection system and personnel in nuclear power plants should be provided.

The licensee's response to the requirement listed above refers to licensee Procedure UNT-050-013, "Fire Protection Program." Procedure UNT-050-013 states that prefire strategy format and content requirements are described in licensee Procedure FP-001-018, "Pre-Fire Strategies, Development and Revision." Procedure FP-001-018, Section 6.1.3 states that:

> Pre-fire strategies should include ... ventilation system operation that ensures desired plant air distribution when the ventilation flow is modified for fire containment or smoke clearing operations.

Contrary to the above requirement, the prefire strategy for vital switchgear Room B (fire zone RAB 8B) did not contain adequate information regarding the doors required to be open to allow the desired ventilation flowpath (Door 11), nor did it contain the required number of smoke ejectors (2) necessary to desmoke the switchgear room in a manner that would allow the implementation of Procedure OP-901-524, "Fire In Areas Affecting Safe Shutdown." Specifically, a manual action, which serves as a compensatory measure for the licensee's

noncompliance with UFSAR Section 9.5.1.3.1.D.1.(a) requirements for separation of safe shutdown trains would not be feasible based on smoke levels in vital switchgear Room AB (fire zone RAB 8C). Switchgear Room AB is located immediately next to switchgear Room B, and a fire in one of the rooms would allow smoke to enter the other room due to a large opening at the top of the wall that separates the switchgear rooms. For the manual action in switchgear Room AB to be feasible, switchgear Room B would need to be desmoked within 1 hour. Per Calculation ECF-05-003, "Manual Action Fire Model - RAB 8," the only way to ensure that switchgear Room AB is habitable within 1 hour is to desmoke switchgear Room B with two smoke ejectors and by ensuring that Door 11 remains open to allow a supply of fresh air to replace the smoke being ejected out to Dry Cooling Tower B area through Door 51.

.2 UFSAR Section 9.5.1.3.1.C.8, "Corrective Action" describes Branch Technical Position APCSB 9.5-1, Revision 0, Appendix A requirement that:

> Measures should be established to assure that conditions adverse to fire protection, such as failures, malfunctions, deficiencies, deviations, defective components, uncontrolled combustible material and nonconformances are promptly identified, reported, and corrected.

The licensee's response to the requirement listed above refers to the Quality Assurance Program Manual (Special Scope) and the Fire Protection Program. Procedure UNT-050-013, "Fire Protection Program," contains no discussion of corrective action criteria. The Quality Assurance Program Manual (Special Scope) Section 5.10, "Corrective Action," refers to Site Directive W2.501, "Corrective Action." Site Directive W2.501, and the corrective action criteria contained therein, was subsumed by Procedure EN-LI-102, "Corrective Action Process." Procedure EN-LI-102, Section 5.8 [2](f) requires, in part, that corrective actions be "timely." Procedure EN-LI-102, Attachment 9.4, "Corrective Action Processing Guidelines," directs that a Category C condition report corrective action should be, "corrected within a timeframe specified by the CRG (normally less than 180 days)."

The noncompliance discussed with UFSAR Section 9.5.1.3.1.B.1 was first identified as a potential vulnerability in Condition Report CR-2006-2407, dated August 21, 2006, and a corrective action recommendation was made to revise the deficient procedure. On August 28, 2006, Condition Report CR-2006-2407 was closed to Condition Report CR-2006-0388 corrective action CA-4, which took action to consider the need to revise the pre-fire plan. On September 5, 2006, Condition Report CR-2006-0388 corrective action CA-5 was created to revise the pre-fire plan. On February 12, 2007, Condition Report CR-2006-0388 corrective action CA-5 was closed to Condition Report CR-2007-0346 corrective action CA-17, which had a due date of December 31, 2008. Contrary to the above corrective action timeliness requirement, the deficient procedure was not corrected until September 14, 2007, after the senior resident inspector discussed the nonconformance with licensee management.

Analysis. The licensee's failure to follow Waterford Steam Electric Station, Unit 3 Facility Operating License Condition 2.C.9 and implement their fire protection program as

described in the UFSAR is a performance deficiency. The finding was more than minor because it was associated with the mitigating systems cornerstone objective (Protection Against External Factors) to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, Appendix F, Phase 1 initial qualitative screening, the issue screened as having very low safety significance because the compensatory manual action required to safely shut down the plant is not needed in order to reach hot shutdown. This finding had a crosscutting aspect in the area of problem identification and resolution. Specifically, the licensee's personnel corrective action process failed to take appropriate corrective actions to address the safety issue in a timely manner (P.1(d)).

Enforcement: Waterford Steam Electric Station, Unit 3 Facility Operating License Condition 2.C.9 requires that the licensee implement their fire protection program as described in the UFSAR. Contrary to the above, the licensee failed to maintain an adequate prefire strategy procedure as described in the UFSAR. Also, contrary to the above, the licensee failed to follow the corrective action program as described in the UFSAR. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program (Condition Report CR-WF3-2007-3264), it is considered a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000382/2007004-03, Inadequate Procedure for a Fire in Vital Switchgear Room B.

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors completed a semiannual trend review of repetitive or closely related issues associated with the Appendix R required emergency lights to identify trends that might indicate the existence of more safety significant issues. The inspectors' review consisted of the 4 year period between January 2003 to September 2007. When warranted, some of the samples expanded beyond those dates to fully assess the issue. The inspectors also reviewed corrective action program items associated with troubleshooting. The inspectors compared and contrasted their results with the results contained in the licensee's quarterly trend reports. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

(Closed) Unresolved Item 05000382/2000007-02: Determine the Qualification of Heymc Fire Wrap as a 1-hour Rated Fire Barrier

a. Inspection Scope

The team had opened this item because of questions related to the acceptability of the tested configuration (4-inch conduits) versus the installed configurations (1- and 2-inch conduits) in Fire Area RAB-2 and indicated further NRC review was required.

An NRC inspector and a senior reactor analyst performed an in-office review of the licensee's interim measures and risk assessment to determine if the licensee had demonstrated that the significance of the issue was less than high safety significance (Red). The inspector performed this inspection by reviewing the documents listed in the attachment and discussed below. The inspector and senior reactor analyst discussed the issues with the fire protection engineer and licensee probabilistic safety assessment personnel.

The inspector performed the evaluation in this manner because Waterford 3 formally committed to converting their Fire Protection Program to comply with the requirements of 10 CFR 50.48(c) and NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition, prior to December 31, 2005. This involves using a risk-informed methodology. The conversion and licensing processes are expected to identify and address a variety of difficult issues that are normally the subject of triennial fire protection inspections. Since any findings in this area will be addressed under the new, rather than the existing, program, the NRC has adapted its inspection and enforcement of certain issues for plants in this situation.

b. Findings

Introduction. The inspector identified an apparent violation of License Condition 2.C.9 because the licensee failed to maintain adequate separation between redundant trains of safe shutdown equipment. Specifically, NRC had determined that the installed Heymc fire barrier material can not provide the required 1 hour of protection. However, this violation will not be cited since the licensee met the Enforcement Policy criteria for enforcement discretion for a plant committed to adopting NFPA Standard 805.

Description. When identified in Calendar Year 2000, the team determined that the installed conduits did not match the tested configuration described in the fire test report for the Heymc fire wrap material. The licensee used the Heymc fire wrap as a 1-hour fire rated barrier to separate safe shutdown functions within the same fire area. The laboratory had performed testing of 4-inch diameter Heymc-wrapped conduits; however, the team identified 1- and 2-inch diameter conduits containing safe-shutdown cables wrapped with Heymc.

The NRC conducted testing of Heymc material and documented the test results in Information Notice 2005-07, "Results of Heymc Electrical Raceway Fire Barrier System Full Scale Fire Testing." NRC tested the following four common methods of joining the Heymc material into a complete electrical raceway fire barrier system: (1) using stitched joints, (2) using minimum 6-inch collars over a joint, (3) using minimum 2-inch overlapping of the mats, and (4) using through bolts with fender washers. The information notice describes the impact upon each of the methods, which resulted in opening of each of the joint systems and exposing the assembly (conduit, cable tray,

junction box and air drop cable) to the furnace environment. The testing demonstrated that all but one assembly (conduit or cable tray) experienced temperatures capable of damaging plant cables as identified in Inspection Manual Chapter 0609, Appendix F, Fire Protection Significance Determination Process, Attachment 7.

In response to Information Notice 2005-007, the licensee initiated compensatory measures and initiated Condition Report 2005-01178. As immediate corrective actions, the licensee initiated hourly fire tours of the 19 fire areas that contained the Heymc material. The licensee determined in their apparent cause evaluation that the presence of sprinklers (except for two areas with approved deviations from Appendix R) and the presence of their fire brigade ensures that a fire would not impact the plants ability to perform a safe shutdown following a fire. As long-term corrective actions, the licensee developed a Heymc Resolution Action Plan that included: identifying the material locations and configurations, testing the configurations, initiating plans and training for replacing the Heymc as needed with an approved fire barrier material, and performing a study to identify options for addressing the Heymc since replacing all of the Heymc was identified as cost prohibitive.

NRC issued Generic Letter 2006-03, "Potentially Nonconforming Heymc and MT Fire Barrier Configurations," to require that licensees evaluate their facilities to confirm compliance with existing regulations. Specifically, Generic Letter 2006-03 required licensees to discuss the installation of Heymc or MT barrier materials and the impact on their facility including whether the installation was described in their licensing basis. The generic letter further required a description of their corrective actions and planned completion date.

The licensee described in their Generic Letter 2006-03 response that they had Heymc installed extensively throughout the facility on conduits, cable trays, containment penetrations, and inside containment as a radiant energy shield. Because of the estimated cost to replace the Heymc, the licensee elected to adopt NFPA 805, in accordance with 10 CFR 50.48(c). The licensee described their intent to adopt NFPA 805 by letter dated December 21, 2005. Because of the time required to transfer to an NFPA 805 based fire protection program, the licensee indicated they would not have all corrective actions completed by December 2008.

Analysis. Failure to meet the separation requirements for a 1-hour fire barrier was a performance deficiency since the licensee did not comply with their Fire Protection Program, as required by License Condition 2.C.9. This finding was more than minor because it affected the protection against external factors attribute of the Mitigating Systems cornerstone. As specified in the enforcement policy, the licensee had performed a simplified risk assessment, included as Attachment B. The inspectors reviewed the simplified fire area-by-fire area risk assessment and determined that the licensee demonstrated that the risk was less than high safety significance (Red).

Enforcement. License Condition 2.C.9, states, in part, that the licensee shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for the facility through Amendment 36 and as approved in the Safety Evaluation Report through Supplement 9. Section 9.5.1.4(1) of the Waterford Safety Evaluation Report states that the licensee committed to provide 1-hour fire rated barriers to protect one division of shutdown-related cables in cable trays

and conduits in certain fire areas. Contrary to the above the licensee had installed an inadequate 1-hour fire barrier in 19 different fire areas that could have impacted the ability to safely shutdown the facility. The licensee included this item in their corrective action program as Condition Report 2005-01178.

Because the licensee committed to adopting NFPA Standard 805 and changing their fire protection program license basis to comply with 10 CFR 50.48.(c), this issue is covered by enforcement discretion in accordance with the NRC Enforcement Policy. Specifically, the licensee: (1) would have identified and addressed this issue during the conversion to NFPA Standard 805, (2) had entered this issue into their corrective action program and implemented appropriate compensatory measures, (3) demonstrated the finding would not be categorized under the Reactor Oversight Process as Red or a Severity Level I violation, and (4) submitted their letter of intent prior to December 31, 2005. The inspector determined that this violation meets the criteria for enforcement discretion for plants in transition to a risk-informed, performance-based fire protection program as allowed per 10 CFR Part 50.48(c). Since all the criteria were met, the NRC is exercising enforcement discretion for this issue.

4OA6 Meetings, Including Exit

Exit Meeting Summary

.1 On August 16, 2007, the inspector presented the occupational radiation safety inspection results to Mr. J. Kowalewski and other members of your staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

.2 On September 20, 2007, the inspectors discussed the results of their review with Mr. O. Pipkins, Senior Licensing Engineer. The inspectors returned all proprietary information to the licensee.

.3 On September 19, 2007, the emergency preparedness inspector conducted a telephonic exit meeting to present the inspection results to Mr. J. Lewis, Manager, Emergency Preparedness, who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

.4 On October 4, 2007, the resident inspectors presented the inspection results to Mr. Joe Kowalewski and other members of licensee management at the conclusion of the inspection. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Anders, Superintendent, Plant Security
H. Brodt, Risk Analyst
K. Cook, Director, Nuclear Safety Assurance
L. Dauzat, Supervisor, Radiation Protection
A. Dodds, Manager, Operations
G. Fey, Planning and Scheduling
J. Kowalewski, General Manager, Entergy
J. Lewis, Manager, Emergency Preparedness
D. Marpe, Project Manager
M. Mason, Technical Specialist, Licensing
C. Miller, Assistant Manager, Radiation Protection
R. Murillo, Manager, Licensing
D. Newman, Supervisor, Radiation Protection
K. Nichols, Director, Engineering
B. Pilutti, Manager, Radiation Protection
R. Putnam, Manager, Programs and Components
S. Ramzy, Engineer, Radiation Protection
G. Scott, Engineer, Licensing
K. T. Walsh, General Manager, Plant Operations

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000382/2007004-01	NCV	Inadequate Boric Acid Leak Evaluations(Section 1R19)
05000382/2007004-02	NCV	Missed Reactor Coolant System Chemistry Samples (Section 1R22)
05000382/2007004-03	NCV	Inadequate Procedure for a Fire in Vital Switchgear Room B (Section 4OA2)
05000382/2007004-04	AV	Determination as to the qualification of Heymc fire wrap as a rated 1-hour fire barrier (Section 4OA5)

Closed

05000382/2007004-01	NCV	Inadequate Boric Acid Leak Evaluations (Section 1R19)
05000382/2007004-02	NCV	Missed Reactor Coolant System Chemistry Samples (Section 1R22)
05000382/2007004-03	NCV	Inadequate Procedure for a Fire in Vital Switchgear Room B (Section 4OA2)
05000382/2000007-02	URI	Determination as to the qualification of Heymc fire wrap as a rated 1-hour fire barrier (Section 4OA5)
05000382/2007004-04	AV	Determination as to the qualification of Heymc fire wrap as a rated 1-hour fire barrier (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R05: Fire Protection (71111.05)

Procedures/Documents

NUMBER	TITLE	REVISION
Administrative Procedure UNT-005-013	Fire Protection Program	9
Operating Procedure 009-004	Fire Protection	11-8
Maintenance Procedure MM-007-010	Fire Extinguisher Inspection and Extinguisher Replacement	13
Administrative Procedure UNT-005-013	Fire Protection Program	9
Fire Protection Procedure FP-001-015	Fire Protection System Impairments	17
Fire Protection Procedure FP-001-017	Transient Combustibles	19
Training Manual Procedure NTP-202	Fire Protection Training	11-4

Section 1R06: Flood Protection Measures (71111.06)

Procedures/Documents

NUMBER	TITLE	REVISION
EN-LI-113	Licensing Basis Document Change Process	1
W3P82-0652	Resolution of Hydrology Branch Concerns	March 30, 1982
OP-100-014	Technical Specification and Technical Requirements Compliance	301
EC-M99-010	Dry Cooling Tower Basin Ponding Analysis	0
DCP-3521	Reroute Dry Cooling Tower Sump Pumps Discharge to Circulating Water System	4
ECP-97-024	Pipe Stress Calculation: Dry Cooling Tower Circulating Water Piping	0

Condition Reports

CR-WF3-2003-0448	CR-WF3-2007-0830	CR-WF3-2007-1695
CR-WF3-2006-0411	CR-WF3-2007-1421	CR-WF3-2007-2574
CR-WF3-2007-0818	CR-WF3-2007-1693	
CR-WF3-2007-0824		

Section 1R12: Maintenance Effectiveness (71111.12)

Procedures/Documents

NUMBER	TITLE	REVISION
DC-121	Maintenance Rule	1
NUMARC 93-01	Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	3
Engineering Report W-SE-2005-001	Waterford 3 Maintenance Rule Periodic (a)(3) Assessment	0

Condition Reports

CR-WF3-2005-4124	CR-WF3-2006-1612	CR-WF3-2007-2704
CR-WF3-2005-4845	CR-WF3-2006-2163	CR-WF3-2007-2988
CR-WF3-2006-0766	CR-WF3-2007-1415	CR-WF3-2007-3168
CR-WF3-2006-1111	CR-WF3-2007-2392	CR-WF3-2007-3169
CR-WF3-2006-1532	CR-WF3-2007-2693	

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

Procedures/Documents

NUMBER	TITLE	REVISION
OP-008-008	Shield Building Ventilation	8
OP-009-008	Safety Injection System	19
OP-009-003	Emergency Feedwater	13
OP-903-127	Reactor Trip Circuit Breaker Post Maintenance Test	3
OI-037-000	Operations Risk Assessment Guideline	2
EN-WM-101	On-Line Work Management Process	1

Drawings

NUMBER	TITLE	REVISION
B-289, Sheet 90	Power Distribution and Motor Data 480V MCC 3A315-S One Line Diagram	8
B-289, Sheet 91	Power Distribution and Motor Data 480V MCC 3A315-S One Line Diagram	10
B-289, Sheet 93	Power Distribution and Motor Data 480V MCC 3B315-S One Line Diagram	8
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1564-318	Seal Oil	15

Section 1R15: Operability Evaluations (71111.15)

Procedures:

NUMBER	TITLE	REVISION
EN-OP-104	Operability Evaluation	1
OP-035-000	Notification Matrix	6

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CR-WF3-2007-2041	CR-WF3-2007-2610	CR-WF3-2007-2690
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Section 1R19: Post Maintenance Testing (71111.19)

Boric Acid Evaluations

04-0064 04-0094 04-0114 06-0341 06-0342 07-0544 07-0545

Condition Reports

CR-WF3-2007-3315 CR-WF3-2007-3390

Drawing

1564-978, "Containment Spray Pump Material Documentation," Revision 11

Procedures

EN-DC-319, "Inspection and Evaluation of Boric Acid Leaks," Revisions 0 and 1

NOECP-107, "Boric Acid Corrosion Control Program," Revision 1

UNT-006-031, "Identification and Evaluation of Boric Acid Leakage," Revision 0

Work Orders

WO-120970	WO-51089006	WO-5108280
WO-50205090	WO-51205387	WO-18881

Section 1R22: Surveillance Testing (71111.22)

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MM-007-010	Fire Extinguisher Inspection and Replacement	15
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OP-903-035	Containment Spray Pump Operability Check	12

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CE-003-327	Operation of the Primary Sample Panel	18
CE-001-004	Periodic Analysis Scheduling Program	301

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WO-51190568	WO-51191844
WO-51191874	WO-51202065

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

Section 2OS2: ALARA Planning and Controls (71121.02)

Corrective Action Documents

2006-3932	2006-4306	2007-0077	2007-0578	2007-0715
2007-0744	2007-0882	2007-0899	2007-0985	2007-1043
2007-1044	2007-1102	2007-1735	2007-2236	

Audits and Self-Assessments
2006 Annual Radiation Protection Report
Monthly Radiation Protection Report, July 2007
Audit Report QA-14-2007-WF3-1, Radiation Protection
Radiation Safety Assessment, LO-WLO-2006-0112-001
Quarterly Roll-Up Assessment, LO-WLO-2006-00124 CA 12, Fourth Quarter 2006
Quarterly Roll-Up Assessment, LO-WLO-2007-00047 CA 12, First Quarter 2007
Focused Assessment of Occupational Radiation Safety WLO-2007-00066
Focused Assessment of Alpha Monitoring Program WLO-2007-00075 CA001

Radiation Work Permits
RWP 2007-0055
RWP 2006-0509
RWP 2006-0510
RWP 2006-0621
RWP 2006-0705
RWP 2005-0717

Procedures
EN-RP-101, “Access Control for Radiologically Controlled Areas,” Revision 2
EN-RP-105, “Radiation Work Permits,” Revision 7
EN-RP-108, "Radiation Protection Posting," Revision 4
EN-RP-122, “Alpha Monitoring,” Revision 0
EN-RP-131, “Air Sampling,” Revision 3

EN-RP-501, "Respiratory Protection Program," Revision 2
HP-001-114, Control of Temporary Shielding," Revision 10

Shielding Requests
Temporary Shielding Request 2006-06
Temporary Shielding Request 2006-08
Temporary Shielding Request 2006-16
Temporary Shielding Request 2006-22

Section 4OA1: Performance Indicator Verification (71151)
Performance Indicator Review Package 1st Quarter 2006
Performance Indicator Review Package 2nd Quarter 2006
Performance Indicator Review Package 3rd Quarter 2006
Performance Indicator Review Package 4th Quarter 2006
Performance Indicator Review Package 1st Quarter 2007
Performance Indicator Review Package 2nd Quarter 2007

Section 4OA2: Identification and Resolution of Problems (71152)

Procedures/Documents

NUMBER	TITLE	REVISIONS
EN-LI-113	Licensing Basis Document Change Process	1
OP-100-014	Technical Specification and Technical Requirements Compliance	301
EN-LI-110	Commitment Management Program	0
UNT-005-013	Fire Protection Program	9
FP-001-020	Fire Emergency / Fire Report	12
FP-001-018	Pre-Fire Strategies, Development	9
W2.501	Corrective Action	8
EN-LI-102	Corrective Action Process	10
EN-LI-118	Root Cause Analysis Process	7
OP-902-009	Standard Appendices	3
EC-F00-026	Appendix R Revalidation Project Post Fire Safe Shutdown	1
QAPM	QAPM Special Scope (Fire Protection)	2
ECF-05-003	Manual Action Fire Model - Fire Area RAB 8	0
OP-901-524	Fire In Areas Affecting Safe Shutdown	2

Condition Reports

CR-WF3-2003-2058
CR-WF3-2003-2735
CR-WF3-2003-2736
CR-WF3-2003-2737
CR-WF3-2003-2738
CR-WF3-2003-2739
CR-WF3-2003-2839
CR-WF3-2003-2840
CR-WF3-2003-2841
CR-WF3-2005-3553
CR-WF3-2006-0213
CR-WF3-2006-0346
CR-WF3-2006-0388
CR-WF3-2006-0954
CR-WF3-2006-0956
CR-WF3-2006-0958
CR-WF3-2006-0960
CR-WF3-2006-2407
CR-WF3-2007-1708
CR-WF3-2007-2591
CR-WF3-2007-2610
CR-WF3-2007-3264

Section 4OA5: Other Activities (71111.05T (OA))

Miscellaneous

Information Notice 2005-07, "Results of Heymc Electrical Raceway Fire Barrier System Full Scale Fire Testing," dated April 1, 2005

Letter CNRO-2005-00064, "Letter of Intent to Adopt NFPA 805 - Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants 2001 Edition," dated December 21, 2005

Generic Letter 2006-03, "Potentially Nonconforming Heymc and MT Fire Barrier Configurations," dated April 10, 2006

Letter W3F1-2006-0028, "Response to Generic Letter 2006-03, Potentially Nonconforming Heymc and MT Fire Barrier Configurations," dated June 7, 2006

Simplified Fire Risk Assessment for Heymc Fire Wrap, dated September 7, 2006

Project Plan for Transition to 10 CFR 50.48(c) NFPA 805 (Waterford 3 version with Heymc replacement plan)

Condition Report 2005-01178

Fire Impairment 05-0334

Updated Final Safety Analysis Report

LIST OF ACRONYMS

NRC	Nuclear Regulatory Commission
PDR	Public Document Room
CFR	Code of Federal Regulations

Simplified Fire Risk Assessment For Hemyc Fire Wrap

Waterford 3

September 7, 2006

Attachment

Purpose

Entergy is requesting enforcement discretion for Hemyc fire wrap inoperability [Ref. 1] during the transitioning of the Waterford 3 fire protection design basis to NFPA 805. One of the conditions for the NRC granting this enforcement discretion is that all of the Hemyc problems are below the *Red* level of risk significance under the NRC's Significance Determination Process [Ref. 2]. This simplified fire risk assessment confirms that all of the Hemyc fire wrap problems are less than Red in risk significance.

An assessment of the Hemyc installations at the Waterford 3 Nuclear Station was performed February 20-24, 2006, by Kleinsorg Group, LLC. The goal of this assessment was 1) to determine whether or not NFPA 805 methodologies could be applied to show that the issues with wrap qualification could be reasonably solved using NFPA 805 Methodologies, and 2) to provide input to Entergy as to the 'color' of the finding to meet enforcement discretion. [Ref. 3] The present analysis is taken from that assessment, with the exception of fire zones RAB 6 and RCB, for which analysis was performed by Waterford 3 staff.

Risk Assessment

Bounding estimates of the risk impact of assuming the Hemyc fire wrap provides no protective function are provided for each of the fire zones which contain Hemyc wrap credited in the Waterford 3 Appendix R Safe Shutdown Analysis [Ref. 5]. Table 1 shows the affected fire zones. With the exception of fire zones RAB-6 and RCB, the risk assessments are reproduced directly from the Kleinsorg Group analysis [Ref. 3].

Table 1 - Fire Zones Containing Safe Shutdown-Credited Appendix R Hemyc Wrap

Fire Zone	***Description***
RAB 1B	Control Room H&V
RAB 2	H&V Mechanical
RAB 3	HVAC Equipment Corridor/Vestibule
RAB 5	Electrical Penetration Area B
RAB 6	Electrical Penetration Area A
RAB 7	Relay Room
RAB 8A	Switchgear Room A

RAB 8B	Switchgear Room B

Fire Zone	***Description***
RAB 8C	Switchgear Room A/B
RAB 17	CCW Heat Exchanger B Rm
RAB 23	Corridors/Common Area
RAB 27	Various Spaces
RAB 31	Corridors and Passageways
RAB 34	Valve Galleries
RAB 36	HP Safety Injection Room
RAB 37	Emergency Feedwater Pump Room
RAB 39	General Area
RCB	Reactor Containment Building

Fire Zone RAB 1B

Fire Zone RAB 1B is the Control Room H&V Equipment Area located on the +46 ft elevation. The targets of interest in this room are two redundant DC power circuits to control room lighting. The "A" Train branch circuit cable is in conduit which is wrapped.

The control lighting is not explicitly modeled in the PRA. A total loss of lighting could affect HEP for operator actions in the main control room. However, the configuration of the room together with the in-situ fire ignition sources indicate fire induced failure of both targets due to a single fire is not a credible event. However, for the purposes of developing a bounding risk characterization the frequency of a fire event that would disable any one train of the lighting circuit is estimated. This estimated frequency is then combined with an assumed random failure probability of 1.0E-3 for the 'undamaged' train of lighting. It is noted that this treatment estimates the total CDF for the scenario rather than just the 'change'. Because of uncertainty related to interactions in the room and the specific role of lighting in the treatment of in control room operator actions, this approach was used to ensure a bounding estimated was obtained.

The fire ignition frequency for an applicable scenario is developed based on the transient fire frequency of 1E-3 per yr for the compartment. This is because the walkdown did not identify any fixed fire ignition sources that represent a credible fire threat to the conduits. A floor based transient package could

impact the cables/conduit individually. The total floor area of the room is 2582 ft^2. To impact the conduits, the fire would need to be within 10 ft. of the conduit. Assuming a 100 sq ft area in which the fire

needs to be gives yields an area factor of 3.9E-2. The combination of these terms results in a bounding CDF of:

3.9E-2 x 1E-3 x 1.0E-3 = 3.9E-8 per yr

Fire Area RAB 2

Fire Area RAB 2 is the H&V Mechanical (Chiller) Room located on the +46 ft elevation. The room contains the essential water chillers (EWC) and HVAC fans and charcoal filters for several air handling units. Hemyc wrapped cable targets associated with the swing (A/B) EWC were identified and are redundant to the A and B EWCs and their associated cables.

The chiller room includes three trains of chillers that provide cooling to the essential service safe shutdown pumps and equipment rooms. The internal events PRA results for WSES shows that the conditional core damage probability (CCDP) is very sensitive to potential fire induced failure of the chillers. Consequently, the spatial interactions within the compartment between the chillers and any other circuitry that affect the Turbine Driven EFW or motor driven AFW pumps would have significant influence on the risk characterization of this compartment. This area also includes some cables associated with the MSIVs. The extent of damage to the equipment and cables from credible fires within the compartment can not be readily determined without a comprehensive review of all cable and the spatial data associated with cable routing and the fixed ignition sources. It is not reasonable within the scope of this review to develop an estimate of the change in CDF due to the removal of the wrap.

It is acknowledged that the enforcement discretion associated with transition to NFPA 805 requires confirmation that conditions consistent with an SDP ‘RED’ finding do not exist. While an estimate of the CDF for this compartment cannot be reasonably established, there is sufficient information to ascertain whether a CDF consistent with a ‘RED’ finding exists. This determination is developed using the NUREG/CR-6850 , Table 6-1 generic plant-wide fire frequency of 7.4E-3 per yr for ventilation subsystems. The application of the ventilation subsystem generic frequency is consistent with the NUREG/CR-6850 guidance for the treatment of chillers. The same table shows that 5% of these fire events are oil fire events. In addition, NUREG/CR-6850, Appendix E, Section E.2 recommends a severity factor of 0.02 for large oil fire events. The calculation of the fire frequency for each chiller requires that the total number of ventilation subsystems in the plant be known. However, since this supplemental assessment is purely for the purposes of establishing whether a ‘RED’ condition exists, this partitioning is ignored. The combination of these three factors results in a bounding and extremely conservative fire initiating event frequency of:

$$7.4E-3 \; x \; 0.05 \; x \; 0.02 = 7.4E-6 \; per \; yr$$

The fire initiating event frequency, by itself, is more than an order of magnitude below the SDP 'RED' threshold of 1E-4 per yr. Because of this, no further assessment is considered necessary to conclude that the risk related criteria associated with enforcement discretion are satisfied.

Fire Area RAB 3

Fire Area RAB 3 is the HVAC Equipment Room Corridor and Vestibule Area located on the +46 ft elevation. The room primarily contains HVAC fans and associated equipment.

The wrapped conduit and the conduit containing redundant circuits for the "B" and "A/B" Train Switchgear Rooms cooling units (AH-25A & B) are routed vertically up the east wall of the room and then across the room in the overhead and out through ceiling near the west wall. At the point of exit through the ceiling of the room, the conduits approach each other but remain at least 24" apart.

There are no fixed ignition sources near the vertical sections of the conduit and fixed sources that are present in the room are not capable of producing damage to the conduits/cables at the ceiling level due to insufficient combustibles to from a ceiling jet or hot layer. Therefore, the only credible fire event that could impact both circuits is a transient that affects both conduit risers. This fire could result in damage to the cables in the conduits and the loss of cooling for the "B" and "A/B" switchgear room cooling units. The loss of room cooling would result in the loss of the "B" and "A/B" Train essential service switchgear. This would result in the loss of the "B" train safety components. However, because of the localized affect of such a fire, an estimate of the associated CCDP can be obtained using the existing internal events PRA results. This estimate was developed assuming a reactor trip (%T1) and loss of AH-25A and AH25B. A review of the cutset file found that fire induced failure of the cooling units can be treated in a bounding fashion by setting the CCF event (UCCCHLRABF) to 'true'. The resultant CCDP is 1.4E-2.

The typical room transient ignition frequency of 1E-3 is assumed. The total floor area for the room is 2985 ft^2. Assuming a 100 sq ft fire, the transient ignition frequency area factor is 3.4E-2.

Combining these values results in a 'bounding' CDF estimate of:

1E-3 x 3.4E-2 x 1.4E-2 = 4.8E-7 per yr

Fire Area RAB 5

Fire Area RAB 5 is the Electrical Penetration Area B located on the +35 ft elevation. The single wrapped conduit of concern within this compartment contains cables for one (224B) of two valves in one of two flow paths to SG2. The cable is part of a 4-20mA loop the failure of which may produce a full open response of the flow control valve or fail the valve closed. It is assumed that a postulated inter-cable hot short that

results in the valve failing 'open' for a sufficient duration to cause irrecoverable conditions is very low and is therefore not explicitly treated. In addition, although not specifically confirmed, it is likely that operators will be able to mitigate the full flow injection and therefore loss of flow is the functional loss of concern. The compartment contains a large quantity of "B" train component cables; the exact functions of which are not known. It is believed that the MSIV cables are located in this room and therefore PCS may be lost for a fire which is severe enough to spread to other trays within the compartment. Based on discussions with plant personnel offsite power cables are not routed through this area. The wrapped conduit is approximately 5 ft above floor level and is located within a small vestibule area.

The CCDP for this scenario is not explicitly calculated. Instead, an assumed CCDP for single train shutdown of 1E-2 is applied. There are no fixed ignition sources in this compartment. Therefore, the credible fire that could impact the circuit and spread to the trays is a floor based transient fire. The typical room transient ignition frequency of 1E-3 is assumed. The total floor area for the room is 5406 ft^2. Assuming a 100 sq ft fire, the transient ignition frequency area factor is 1.85E-2.

Combining these values of damage change and likelihood of a fire damaging the wrapped and un-wrapped conduits concurrently:

1E-3 x 1.85E-2 x 1.0E-2 = 1.9E-7 per yr

Fire Area RAB 6

Fire Area RAB 6 is the Electrical Penetration Area A located on the +35 ft elevation.

The scenario of concern is fire-induced failure of CCW. Cables associated with the wet and dry cooling fans for Train A of CCW (i.e., the heat sink for Train A) are in this fire area, as are cables associated with the swing (AB) CCW pump. CCW pump B cables are protected by Hemyc fire wrap. Thus, a fire in RAB 6 could be hypothesized to fail both CCW trains. This would cause loss of cooling of the reactor coolant pumps (RCPs), with a resulting manual reactor trip. If an RCP seal failed, a small LOCA would be induced. Without CCW, containment heat removal would be unavailable (both containment spray and the containment fan coolers are dependent on CCW cooling) and safety injection could fail in the recirculation mode due to loss of required suction head.

The CDF for this scenario is estimated as follows. The probability of an RCP seal failure given loss of CCW is 7.8E-5 from the Waterford 3 PSA. This is the total probability of any one of the 4 pumps failing. Assuming the typical room transient ignition frequency of 1E-3 from RAB 5, but conservatively ignoring the area adjustment, gives a CDF of:

1E-3 x 7.8E-5 = 7.8E-8 per yr

Fire Area RAB 7

Fire Area RAB 7 is collectively the Relay Rooms and is located on the +35 ft elevation. The fire area contains a large number of cable trays, conduit, and electronic equipment such as relays, SUPS, control panels, and battery chargers. Electronic equipment in the compartment is sealed and un-vented. Fire Area RAB 7 is about 100 ft long, 25 ft wide, and has a ceiling that is about 11 ft above the floor. The area is subdivided into four fire zones: RAB 7A, RAB7B, RAB 7C, and RAB7D, each separated from one another by part height concrete block walls. Cable trays generally are either wrapped with Hemyc or are fully covered.

There is about 225 linear feet of credited Hemyc wrap in this fire area protecting one conduit (39347–SA–4) and one cable tray (C203–SA). The conduit is located in Fire Zones RAB 7A, RAB 7B, and RAB 7C near the east wall about 6 ft above the floor. The cable tray is located in Fire Zone RAB 7A and RAB 7C near the east wall and above the relay equipment in Fire Zone RAB 7C about 6 ft above the floor. The conduit and cable tray are redundant to multiple Train B cables and equipment in the area, thus the Hemyc wrapped circuits are essentially single cable targets.

This compartment, consisting of several sub-compartments, includes relay cabinet of all three divisions and serves as a cable spreading area. The specific wrapped conduit of concern includes cables associated with CCW, SD Cooling and MS-ADVs. The wrapped trays of concern include cables of both "A" and "B" train initiation logic. There is a large amount of other cabling routed in the low overhead of the room and many fixed ignition sources. The extent of damage to the equipment and cables from credible fires within the compartment cannot be readily determined without a comprehensive review of circuit analysis and spatial data. The possible interaction of fire source cabinets and target cables are numerous. It is not reasonably within the scope of this review to determine the resultant or change in CDF due to the removal of the wrap.

It is acknowledged that the enforcement discretion associated with transition to NFPA 805 requires confirmation that conditions consistent with an SDP ‘RED’ finding does not exist. While an estimate of the CDF for this compartment cannot be reasonable established, there is sufficient information to ascertain whether a CDF consistent with a ‘RED’ finding exists. This determination is developed using the NUREG/CR-6850, Table 6-1 generic plant-wide fire frequency of 4.5E-2 per yr. for electrical cabinets. The incremental fire frequency associated with transient fires is excluded from this assessment, but is judged to be bounded by the conservative treatment of fire frequency. The calculation of the fire frequency for this compartment requires that the total number of electrical cabinets in the plant and the number representing ‘threats’ to the Hemyc wrapped circuits be known. Because this supplemental assessment is purely for the purposes of establishing whether a ‘RED’ condition exists, a detailed accounting of panel inventories is not required. Instead, this assessment is based on an assumed electrical panel population of 500, a count of 2 panels that represent a threat to the Hemyc wrapped conduits, and a manual suppression credit of 0.10 based

on the installed smoke detectors inside these cabinets. The combination of these factors results in a screening fire initiating event frequency of:

$$4.5E-2 \; x \; {}_{500}2 \; x \; 0.10 = 1.8E-5 \; per \; yr$$

The fire initiating event frequency, by itself, is almost an order of magnitude below the SDP 'RED' threshold of 1E-4 per yr. In addition, the treatment of the postulated fire scenario does not apply a fire severity factor to account for the heat release rate that would be required to represent a credible threat to the Hemyc wrapped raceways above the cabinets. Because of this, no further assessment is considered necessary to conclude that the risk related criteria associated with enforcement discretion are satisfied.

Fire Zone RAB 8A

Fire Zone RAB 8A is the Switchgear Room A located on the +21 ft elevation. The room primarily contains A train switchgear, MCCs, multiplexers, and associated cables and bus ducts.

The wrapped conduit contains the power cable to the "B" and "A/B" Train Switchgear Room air cooling unit AH-25B. The redundant air cooling unit AH-25A for the theses switchgear rooms is also routed through this area and is in fact powered from a bus within the room. Loss of both of these cables would result in loss of cooling for the B and A/B Switchgear rooms. A total loss of the "A" Train switchgear room would result in loss of cooling to the "B" and "AB" switchgear rooms that would cause a total loss of the 4kV essential service buses.

A single floor based transient could impact the conduit (but not the redundant conduit at the same time). The panels directly below the conduit includes DC distribution Panels PDP-87A and -87B and MCC 311A. Review of the MCC loads indicates that a significant loss of event mitigation capability would not occur. These are low voltage panels as are the panels in the vicinity of the wrapped conduit. The conduit is routed over a 4kv busduct connecting the "A" 4 kV SWGR to the "A/B" SWGR. However, a HEAF is not postulated on this duct due to its location in the distribution system downstream of offsite power supplies. An arc on the bus or fault due to smoke accumulation may cause isolation of the bus section but, due to breakers on both ends of the bus and the availability of the opposite train power to the bus, a fire will not be sustained and a long-term loss of power is not assumed.

Based on the walkdown results and the summary of fire modeling expectations, it is anticipated that a credible fire scenario that results in damage to both redundant targets will not be identified. This effectively translates to a condition where the risk results would be the same whether the conduit was wrapped or not. As such, there is no measurable change in CDF or LERF for this compartment.

Fire Zone RAB 8B

Fire Zone RAB 8B is the Switchgear Room B located on the +21 ft elevation. The room primarily contains B train switchgear, MCCs, multiplexers, and associated cables and bus ducts.

The wrapped conduit contains the power cable to the Emergency Service "A" Train Chiller that serves "A" Train equipment and room coolers. The room contains the “B” Train essential switchgear. The panels directly below the conduit includes DC distribution panel DC-EPDPB-DC, MCC 312B, and the CRD cabinets. Review of the loads on these panels indicates that no significant event mitigation systems will be impacted. The conduit is not routed in the vicinity of 4kV SWGR and therefore an HEAF is not postulated. A transient fire or a fire within the panels will not result in the loss of offsite power nor impact PCS systems. Fires within the panels are assumed to remain within the panels due to sealing of all cable entries and the absence of vents in the cabinets. A severe fire is assumed to generate enough heat to cause the panel to have sufficient radiant heat to damage the nearby conduit. The trays in the area include both top and bottom covers, with the exception of one tray’s cables that enter the MCC.

The walkdown of this compartment and the review of credible fire scenarios did not identify a case where redundant features would be adversely affected by a single fire event. This effectively translates to a condition where the risk results would be the same whether the conduit was wrapped or not. As such, there is no measurable change in CDF or LERF for this compartment.

Fire Zone RAB 8C

Fire Zone RAB 8C is the Switchgear Room A/B located on the +21 ft elevation. The room primarily contains switchgear, MCCs, multiplexers, and associated cables and bus ducts associated with both the A/B trains.

This room includes the “A/B” Train switchgear. The wrapped conduits and trays carry cabling for the “A” Train Chiller, HVAC, and several cables which enter the “B” Train aux panel on the floor above. There is a large amount of other cabling routed in the low overhead of the room and a fixed ignition source directly below the tray (MCCAB312). In addition, a floor based transient fire could impact the wrapped target tray and conduit. The extent of damage to the equipment and cables from credible fires within the compartment can not be readily determined without a comprehensive review of circuit analysis and spatial data. It is not reasonably within the scope of this review to determine the resultant or change in CDF due to the removal of the wrap.

It is acknowledged that the enforcement discretion associated with transition to NFPA 805 requires confirmation that conditions consistent with an SDP ‘RED’ finding does not exist. While an estimate of the CDF for this compartment cannot be reasonably established, there is sufficient information to ascertain whether a CDF consistent with a ‘RED’ finding exists. This determination is developed using the NUREG/CR-6850, Table 6-1 generic plant-wide fire frequency of 4.5E-2 per yr. for electrical cabinets. The incremental fire frequency associated with transient fires is excluded from the assessment of electrical cabinet based fires, but is judged to be bounded by the

conservative treatment of fire frequency. In the case of the region above the battery rooms, the only credible fire threat is a transient based fire.

The calculation of the electrical cabinet fire frequency for this compartment requires that their total number in the plant and the number representing 'threats' to the Hemyc wrapped circuits be known. Because this supplemental assessment is purely for the purposes of establishing whether a 'RED' condition exists, a detailed accounting of panel inventories is not required. Instead, this assessment is based on an assumed electrical panel population of 500, a count of 5 panels that represent a threat to the Hemyc wrapped conduits. The walkdown of this compartment did note that electrical cabinets were generally sealed such that only those fire events that have some minimum heat release rate would be capable of challenging the circuits wrapped with Hemyc. While the Hemyc is not 'qualified' for the required fire duration, the testing results do suggest that some residual protection is afforded given the specific source vs. target interaction in this compartment. While a specific treatment of this interaction was not performed, a combined factor of 0.10 is applied to represent the combination of fire severity and manual fire suppression failure. The combination of these factors results in a screening fire initiating event frequency for electrical cabinets of:

$$4.5E-2 \; x \; {}_{500}5 \; x \; 0.10 = 4.5E-5 \; per \; yr$$

The potential fire scenario frequency associated with electrical cabinets must be increased by the incremental fire frequency associated with potential transient based fires in the region above the battery rooms. This fire frequency is estimated using the same parameters used earlier for the treatment of such fires. This treatment results in a fire frequency of 1E-5 per yr.

The cumulative fire initiating event frequency, by itself, is below the SDP 'RED' threshold of 1E-4 per yr. The estimate developed above is conservative in that it does not credit any residual CCDP given a postulated fire event. It is anticipated that postulated fire induced loss of the chiller function can be mitigated by various recovery actions that explicitly addressed in the post fire safe shutdown analysis. Because the failure consequence involves a loss of cooling, it is anticipate the some time would be available for such recovery actions. The application of a screening HEP of 0.10 for actions such as opening of doors or establishment of other cooling means would reduce the fire scenario estimate to well below the 'RED' threshold. Based on the screening results developed no further assessment is considered necessary to conclude that the risk related criteria associated with enforcement discretion are satisfied.

Fire Area RAB 17

Fire Zone RAB 17 is the Component Cooling Water Heat Exchanger B Room located on the +21 ft elevation.

The wrapped trays and conduits of interest are located in the overhead. Intervening trays are present between the wrapped trays and the floor level. The conduits are connected to the wrapped trays or routed close to the room ceiling. The protected cables are for the opposite train CCW, the "A" Train EFW and Chiller B as well as cooling towers. Loss of these cables is assumed to disable both trains of the CCW system. No MSIV cables, offsite power, or other plant BOP systems such as PCS are routed through the compartment based on discussion with plant personnel.

To bound this case, the turbine trip initiator with the loss of CCW imitator is used. The resulting CCDP is 5.74E-5. However, there is large uncertainty associated with the presence of other potentially credited circuits in the vicinity of the wrapped trays that are relatively close to floor level. Therefore, a higher CCDP estimate of 1E-3 is applied based on the assumption that localized fires could damage more than just CCW and the chiller functions.

The fire ignition frequency for an applicable scenario is developed based on the transient fire frequency of 1E-3 per yr for the compartment. The total floor area of the room is 820 ft^2. Assuming a 100 sq ft area yields an area factor of 1.2E-1. Combination of these terms results in a bounding CDF of:

1.2E-1 x 1.0E-3 x 1.0E-3 = 1.2E-7 per yr

Fire Area RAB 23

Fire Area RAB 23 is the Corridor and Common Passageways on the +21 ft elevation.

The wrapped trays and conduits of interest are located in the overhead. The enclosed cables are related to the “B” Train EDG connecting to the “B” Train 4kV essential switchgear. Loss of these cables is assumed to disable the “B” train EDG.

The BOP systems including PCS are assumed to remain available as is offsite power. Assuming a reactor trip with offsite power and PCS available, and the “B” Train EDG failed; the CCDP is 4.63E-6.

The fire ignition frequency for an applicable scenario is developed based on the transient fire frequency of 1E-3 per yr for the compartment. This is because the walkdown did not identify any fixed fire ignition sources that represent a credible fire threat to the conduits. A floor based transient package could impact the cables/conduit. The total floor area of the room is 7,239 ft^2. Assuming a 100 sq ft area yields an area factor of 1.4E-2. Combination of these terms results in a bounding CDF of:

1.4E-2 x 1.00E-3 x 4.63E-6 << 1.0E-9 per yr

Fire Area RAB 27

Fire Area RAB 27 consists of various spaces on the +7 ft elevation, dominated by office type occupancies but also with an electrical equipment room and an HVAC room.

The wrapped conduits of concern are above a false ceiling in the area. There is a large quantity of cabling exiting the switchgear from the room above into this space. It is not possible to determine the potential fires that could impact the raceways of concern nor what other cables would be impacted. The large area below the falsie ceiling is used as office space and contain significant combustible load. The response of the above and below ceiling suppression systems may reduce impacts, however, insufficient spatial and impacted circuits information is available to quantify the risk associated a fire in this area as part of this scope of work.

It is acknowledged that the enforcement discretion associated with transition to NFPA 805 requires confirmation that conditions consistent with an SDP 'RED' finding does not exist. While an estimate of the CDF for this compartment cannot be reasonable established, there is sufficient information to ascertain whether a CDF consistent with a 'RED' finding exists. This determination is developed based on a transient combustible fire frequency of 1.0E-3 per yr. This value is very conservative given the generic values provided in NUREG/CR-6850, Table 6-1. The area is predominately an office area and is therefore provided with an automatic fire suppression system. The failure probability of this suppression system is 2E-2 based on NUREG/CR-6850, Appendix P. The combination of these two factors results in a screening fire frequency of 2E-5 per yr. This value can be further reduced by applying a factor to reflect the relative projected area of the Hemyc area as a fraction of the total available area, and to credit fire brigade response. However, since the developed screening fire initiating event frequency, by itself, is below the SDP 'RED' threshold of 1E-4 per yr. no further assessment is considered necessary to conclude that the risk related criteria associated with enforcement discretion are satisfied.

Fire Area RAB 31

Fire Zone RAB 31 is the Corridor and Passageways at the -4 ft elevation.

The wrapped trays of concern provide for cables to the essential safety equipment in the floors below. The redundant cables are located in a tray riser. There are no fixed sources in this room and no flammable liquids (e.g., oil) are present. No single transient could impact both tray risers. The credible fire scenario for damage to the wrapped cable trays and the enclosed cable is a floor based transient. Given the room size and lack of combustibles, a hot gas layer or ceiling jet capable of impacting both groups of tray risers is not anticipated.

The walkdown of this compartment and the review of credible fire scenarios did not identify a case where redundant features would be adversely affected by a single fire event. This effectively translates to a condition where the risk results would be the same whether the conduit was wrapped or not. As such, there is no measurable change in CDF or LERF for this compartment.

Fire Area RAB 34

Fire Zone RAB 34 is the A and B Valve Galleries located on the -15 ft elevation.

The wrapped conduit of concern is for the LPSI Pump B. This conduit is routed through this compartment to the adjacent pump room as is the conduit to the LPSI A pump.

Both conduits are relatively close to the floor level such that a floor based transient fire could impact the conduits. The conduits are routed over or adjacent to electrical cabinets, therefore there is a potential that a cabinet fire resulting in a “hot box” impact to the cables is possible. However, the redundant conduits are not routed over or adjacent to any single cabinet that could impact both conduits simultaneously. There is a half height wall with wire screen atop dividing the compartment into two sub-compartments and thusly separating the two conduits. Therefore it is not credible for a single floor based fire to impact both conduits. The LPSI pumps are relied upon to mitigate a large LOCA event and for shutdown cooling. A review of the cutsets from the internal events PRA solution found that LPSI related failures were potentially risk significant only given a loss of RCS boundary condition. Because the fire event is not predicted to result in or cause a LOCA or loss of RCS pressure integrity, there is not expected to be any measurable impact to fire related CDF or LERF. Based on this assessment, the ¨CDF and ¨LERF for this configuration is judged to be negligible.

Fire Area RAB 36

Fire Zone RAB 36 is the H.P. Safety Injection Pump A Room located on the -35 ft elevation.

The wrapped conduit of concern is for the cooling unit in the adjacent HP Safety Injection Pump (”B”) Room. The conduits are routed through the overhead in this room well above floor level. The conduits are routed over the HPI “A/B” pump. The HPI pumps contain an unknown quantity of oil. It is not likely that a floor based transient fire could impact these conduits. However, a severe pump fire in the HPI A/B pump directly impacting the conduits and other pump fires developing a hot gas layer could result in damage to the cables.

The pumps in this room and the adjacent B train room are HPSI, Containment Spray, LPSI and the reactor drain tank (B room only). These pumps are relied upon to mitigate a LOCA events and for shutdown cooling. A pump fire or transient fire in this room may disable pumps in this room as a direct result if the fire and in the adjacent “B” train due to loss of cooling resulting from damage to the cables routed through the “A” Train room. Given the equipment in this area and the adjacent compartment a fire is not likely to initiate a trip.

Because the fire event is not predicted to result in or cause a LOCA or loss of RCS pressure integrity, there is not expected to be any measurable impact to fire related CDF or LERF. Based on this assessment, the ¨CDF and ¨LERF for this configuration is judged to be negligible.

Fire Area RAB 37

Fire Zone RAB 37 is the Emergency Feedwater Pump (EFW) A Room located on the -35 ft elevation.

This compartment contains the “A” train EFW pump. The fire wrapped trays located in this zone approximately 15’ above the floor level and trays are solid bottoms. The circuits in the wrapped trays are associated with the EFW Pump “B” Train and may contain cables for the “B” Train charging pump and the turbine drive “A/B” Train EFW

pump. Therefore, the postulated extent of fire induced failures was characterized as being limited by the loss of all EFW. Discussions with plant personnel discussions indicated that there are no circuits whose fire induced failures would challenge the RCS pressure boundary, no MSIV cables and no PCS cables. The postulated consequence of a fire that results in loss of EFW pumps was estimated using a preliminary set of quantification results from the internal events PRA. The CDF given a fire in this location can be estimated by assuming a turbine trip and loss of all EFW simulated by the use of EFW common cause EFW flow blockage BE (QCCALL4CIN). The resulting CCDP is based primarily on PCS and at least one remaining charging pump is about 1.4E-03.

The corresponding scenario frequency is developed from the fire ignition frequency for the single AFW pump times the conditional probability that the fire is large (severe). A floor based transient is not expected to have an impact on the tray due to its height above the floor.

Using NUREG/CR-6850, Table 6-1 lists a generic plant-wide fire frequency of 2.1E-2 per yr. for pump fires. The same table shows that 46% of these fire events are oil fire events. In addition, NUREG/CR-6850, Appendix E, Section E.2 recommends a severity factor of 0.02 for large oil fire events. The calculation of the fire frequency for each Makeup Pump requires that the total number of pumps in the plant be known. For the purposes of this assessment, an estimate of 50 pumps is used. The combination of these four factors results in a single EFW pump fire scenario frequency of:

$$2.1E-2\ x\ 0.46\ x\ 0.02\ x\ {}_{50}1 = 3.9E-6\ per\ yr$$

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The combination of the fire scenario frequency and the CCDP estimate results in a CDF estimate of 5.4E-9 per yr.

Fire Area RAB 39

Fire Zone RAB 39 is the General Area on the -35 ft elevation.

The wrapped trays and conduits are predominately in the overhead; are located away from fixed ignition sources or redundant cable is sufficiently separated such that a floor based transient does not results in damage to the cable or the redundant cable simultaneously. The exception is the pump area that includes the TDFW, CCW booster and charging pumps. It is assumed that the quantity of oil in the oil separator is not significant that there is no credible ignition source for this to be a credible fire threat. In this pump area, the wrapped conduits and trays of concern carry cables for the cables for all trains of EFW. The fixed sources in this area are the TDFW and CCW pumps. The size of the CCW pumps and relative height above floor of the cables results in these fixed source not being a threat. However, the TDFW pump severe fire could potentially damage all EFW trains similar to the worst-case scenario of Compartment RAB-37. A typical transient fire would not be a threat to the wrapped cables. However, a dress out area estimated very conservatively as approximately 15 ft by 20 ft is located under the wrapped “B” Train cables and the “A” train raceways. Like the TDFW pump

fire, this fire potentially impacts the three trains of EFW. As developed for Compartment RAB-37, the loss of the three EFW trains with a rector trip results in a CCDP of 1.4E-3.

The single EFW pump severe fire frequency of 3.9E-6 per yr as developed for Compartment RAB-37 is applicable for this scenario. In addition, a bounding transient fire frequency for the dress out materials storage area is the typical transient frequency of 1E-3 per yr proportioned by an area factor of 1.86E-2 (300/10107). Combining the transient ignition frequency factors yields a frequency of 1.86E-5 per yr.

Combining the fire frequencies and CCDP estimates results in a bounding CDF estimate of:

(1.86E-5 + 3.9E-6) x 1.4E-3 = 3.2E-8 per yr

With a more typical transient fire of 100 ft2 (achieved by removal of the dress out area) the bounding CDF would be:

(6.21E-6+ 3.9E-6) x 1.4E-3 = 1.42E-8 per yr

Fire Zone RCB

Fire Zone RCB is the Reactor Containment Building General Area on the –4 ft, +21 ft, and +35 ft elevations. The Hemyc fire wrap is used as a radiant energy barrier for Shutdown Cooling (SDC) suction isolation valves SI-401A, 401B, 405A, and 405B. These valves form the high pressure-low pressure interface between the Reactor Coolant System and the SDC system. If both SDC isolation valves on a particular SDC line (i.e., 401A and 405A or 401B and 405B) were to fail open, an interfacing system loss of cooling accident (ISLOCA) could result. This potential is minimized by the procedural requirement to remove power from the SI-401A and 401B (motor-operated) valves during operation (OP-009-005 [Ref. 4]).

The Safe Shutdown Analysis [Ref. 5] includes the following evaluation of the effect of an RCB fire on the high-low pressure interface provided by these valves:

> For the high to low pressure interface protected by these valves, a review of the cable routing and associated raceways indicates that there is no source for the 3-phase hot shorts. For example, cables for SI-401A, SI-331A, SI-332A, and the hydraulic motor for SI-405A are all contained in penetration box B30370-SA. However, none of the power cables are normally energized, and none can be spuriously energized by a fire in containment. The entire run of cables for SI-401A(B) and SI-405A(B) is in conduit and junction boxes.

The provides further assurance that fire-induced ISLOCA in these lines is unlikely.

The ISLOCA model included in the Waterford 3 internal events PSA can be used to give a bounding estimate of ISLOCA risk due to a fire in the RCB. The ISLOCA cut sets for the SDC suction lines were used to estimate a conditional ISLOCA probability assuming failure of the SI-405 valves due to a fire. Since the SI-405 valves are hydraulic-operated valves and do not have 480 V power supply breakers that are procedurally opened during operation, they are conservatively assumed to both fail in

an RCB fire. On the other hand, since the SI-401 valves are deenergized, they are assumed not to be susceptible to fire-induced spurious opening.

The calculation of the conditional ISLOCA probability given fire-induced failure of the SI-405 valves was done by setting the SI-405A and B basic events in the cut sets to TRUE, with a resulting probability of 4.0E-4. These modified cut sets include failures of the SI-401 valves due to valve rupture leading to ISLOCA, as well as SI-401 valve leakage events plus failure of the associated outside containment, low pressure isolation valve, SI-407A or B (Note 1). Other SDC suction line ISLOCA scenarios are very low in probability (<1E-10) and are neglected (truncated). Failure of either SDC suction line is included in the cut sets, thus giving the total probability of a fire-induced ISLOCA.

From NUREG/CR-6850 [Ref. 6], the annual frequency of a fire in the RCB that could affect the SI-405 valves is 2.0E-3 (Table C-3; only the transients and hot work initiator is used, because the SI raceways are not located in the vicinity of the Reactor Coolant Pumps). The product of the initiator frequency and the conditional probability of ISLOCA gives the bounding fire-induced ISLOCA frequency without credit for the Hemyc fire wrap. Neglecting credit for operator action to isolate the ISLOCA and the conditional probability of low pressure pipe rupture given failure of the isolation valves make the ISLOCA frequency equal to CDF. The bounding fire-induced ISLOCA CDF is:

2.0E-3 x 4.0E-4 = 8.0E-7 per yr

Note 1: In the case of SI-405A or B leakage, it is assumed that the SDC suction relief valve, SI-406A or B, which has a very large flow capacity, will prevent overpressurizing the outside containment isolation valve, SI-407A or B.

Conclusion

The conservative risk assessment of the impact of Hemyc fire wrap inoperability during the transitioning of the Waterford 3 fire protection design basis to NFPA 805 shows that in all cases the risk associated with the Hemyc fire wrap problem is well less than a significance level of Red in NRC's Reactor Oversight Process Significance

Determination Process (< 1.0E-4 per yr). Waterford 3 therefore meets this NRC criterion for acceptability of enforcement discretion.

References

1. CR-W3-2005-1178.

2. NRC Interim Enforcement Policy (69FR33684, June 16, 2004).

3. "Fire Risk Assessment, Waterford 3 Hemyc Installations, Fire Modeling & Risk Considerations." Kleinsorg Group, LLC, March 2006.

4. OP-009-005, "System Operating Procedure, Shutdown Cooling." Revision 17.

5. ECF00-026, Rev. 1.

6. NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology." August 2005.