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Donna Jacobs
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Waterford 3

W3F1-2012-0064

September 27, 2012

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Response to Request for Additional Information Regarding Adoption of National Fire Protection Association Standard NFPA 805 License Amendment Request
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES:

1. Entergy letter W3F1-2011-0074 "License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition)", Waterford Steam Electric Station, Unit 3 dated November 17, 2011
2. Entergy letter W3F1-2012-0005 "Supplemental Information in Support of the NRC Acceptance Review of Waterford 3 License Amendment Request to Adopt NFPA 805 Waterford Steam Electric Station, Unit 3" dated January 26, 2012
3. NRC Transmittal to Entergy dated July 18, 2012, "Request for Additional Information Regarding Adoption of National Fire Protection Association Standard NFPA 805 (TAC No. ME7602)"

Dear Sir or Madam:

In letter dated November 17, 2011, as supplemented by letter dated January 26, 2012 (References 1 and 2), Entergy Operations, Inc. (Entergy) submitted a License Amendment Request (LAR) for Waterford Steam Electric Station, Unit 3 (Waterford 3) to adopt a new, risk informed - performance based, fire protection licensing basis under 10CFR50.48(c).

In letter dated July 18, 2012 (Reference 3), the NRC staff made a Request for Additional Information (RAI) needed to complete its review which included a set of questions due in 60 calendar days and a set of questions due in 90 calendar days. Attachment 1 provides the responses to those questions considered as the NRC's 60 calendar day RAI's, based on communications on August 23, September 12, and September 25, 2012, requesting various extensions of 60 day responses.

There are no new commitments contained in this submittal.

If you require additional information, please contact the acting Licensing Manager, Michael E. Mason, at 504.739.6673.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 27, 2012.

Sincerely,

DJ/AJH

Attachments:

1. Additional Information (60 day responses) in Support of NRC Review for Waterford 3 NFPA 805 License Amendment Application

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Attachment 1 to

W3F1-2012-0064

Additional Information (60 day responses) in Support of NRC Review for Waterford 3 NFPA 805 License Amendment Application

Additional Information (60 day responses) in Support of NRC Review for Waterford 3 NFPA 805 License Amendment Application

Additional information was requested by the NRC Staff on July 18, 2012 in support of the Review for Waterford Steam Electric Station, Unit 3 (Waterford 3) License Amendment Request (LAR) dated November 17, 2011. The following provides the additional information requested of Waterford 3 by the NRC staff (60 day responses).

PROBABILISTIC RISK ASSESSMENT (PRA)

RAI PRA 01 – 90 Day Response

RAI PRA 02 – 90 Day Response

RAI PRA 03 – 90 Day Response

RAI PRA 04

Please describe the methodology that was used to evaluate defense-in-depth (DID) and the methodology that was used to evaluate safety margins. The description should include what was evaluated, how the evaluations were performed, and what, if any, actions or changes to the plant or procedures were taken to maintain the philosophy of DID or sufficient safety margins.

Waterford 3 Response

Describe the methodology that was used to evaluate defense-in-depth:

The methodology and guidance used to evaluate defense in depth for W3 originated from industry documents NEI-00-01 and NEI-04-02. Following this guidance, Fire Areas were reviewed against the criteria in NFPA 805 section 4.2.3 and were transitioned either deterministically if they met section 4.2.3, or were transitioned as performance-based. Those areas transitioning deterministically were considered to meet the defense-in-depth. For those areas transitioning under section 4.2.4, a Fire Risk Evaluation (FRE) was performed. The FRE considered the risk profile of each fire area based on the ignition frequency (IF), conditional core damage probability (CCDP) and high risk scenarios in each area, and determined if additional defense in depth was warranted for the fire area. This determination was made based on reviewing the numeric value of the IF and CCDP in relationship to the risk importance of the fire area. If either the IF or CCDP dominated the risk profile of a high risk area, then DID actions were considered required and those are documented in Attachment C and Table 4-3 of the LAR.

This review is documented in the fire risk evaluations. During the process, if deficiencies were discovered that did not allow the defense in depth goal to be met, a modification or programmatic change would be considered, documented and can be found in Attachment C and Attachment S of the Waterford LAR submittal.

Describe the methodology that was used to evaluate safety margins:

Both deterministic and probabilistic methods were used to assess the safety margin. In all performance based cases, the PRA model provided the guidance necessary to assure that;

1.) Codes and standards or their alternatives accepted for use by the USNRC were met AND
2.) Safety Analysis acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met or provide sufficient margin to account for analytical and data uncertainties.

The probabilistic methods allow for an integrated approach to assess the facility response to a fire. From this assessment it is possible to define and rank areas of lesser performance that can then be assessed in more detail to determine if improvements are needed. The results of this process are documented in Attachment W of the LAR, with identified actions or changes to plant included in Attachment S of the LAR.

The deterministic methods relied on the elements of the NFPA 805 process related to compliance with applicable codes and standards, thus supporting the maintenance of safety margins. The results of the deterministic process are documented in Attachments A and B of the LAR, with identified actions or changes to plant included in Attachment S of the LAR.

RAI PRA 05

The Transition Report of the LAR summarizes safe and stable conditions but provides limited information about how long the facility can easily maintain hot shutdown (e.g., initial coping time of 24 hours) after which, actions to realign systems and/or resupply required equipment become necessary to maintain safe and stable. Please provide a discussion describing how long safe and stable can be maintained and the actions necessary during and beyond an initial coping time to maintain safe and stable conditions beyond the initial coping time hours such as refilling fluid tanks or re-aligning systems. Please evaluate quantitatively or qualitatively the risk associated with the failure of actions and equipment necessary to extend safe and stable beyond the coping time given the post-fire scenarios during which they may be required.

Waterford 3 Response

From the LAR, Section 4.2.1.2, the 'At Power' safe and stable strategy includes entry into hot standby (Mode 3) and stops prior to the point of manually initiating a cool-down. This is consistent with the Fire PRA (FPRA), which is based on maintaining hot standby for 24 hours. The primary means of maintaining hot standby for both NFPA 805 and the FPRA is maintenance of Emergency Feedwater (EFW) in operation, which requires AC and DC power and a suction water source. In addition, the NFPA 805 definition of safe and stable includes maintenance of Reactor Coolant System (RCS) inventory, Spent Fuel Pool (SFP) inventory maintenance, and availability of at least one train of vital auxiliaries (Component Cooling Water, HVAC, vital electrical power, and Chemical and Volume Control System).

The LAR describes the capabilities and capacities of these systems as needed for maintaining safe and stable conditions in Mode 3. The below discussion describes capabilities to maintain conditions beyond the initial coping time.

Maintenance of Safe and Stable:

Electrical Power

AC power is normally supplied post-trip from offsite power, via the Startup Transformers. If offsite power were lost as a result of the fire or a non-fire-related failure, the Emergency Diesel Generators (EDGs) would start and provide AC power. The EDGs have a 7 day fuel supply for a loss of offsite power with a limiting design basis accident. For a fire event, the load on the EDGs would be significantly less (e.g., no Low Pressure Safety Injection, no Containment Spray), so the fuel supply would last significantly longer than 7 days, giving ample time for replenishing the fuel tanks. Vendor supply of fuel oil is readily available. Therefore, the EDGs can be supplied with fuel indefinitely.

With AC power available long-term via either offsite power or the EDGs, DC power is also available long-term, since the DC buses are normally powered by battery chargers supplied by AC power.

RCS Inventory

Plant-specific thermal-hydraulic (T/H) calculations performed for Waterford 3 in support of post-fire safe shutdown analysis show that maintenance of RCS inventory for 7 days post-fire requires 10,458 gal of borated water for the limiting case. This is much less than the 499,100 gallons of available borated water (2 BAMTs @ 11,800 gal + 475,500 gal in the RWSP) and ensures that RCS inventory can be maintained for many days.

EFW Flow Control Valves

The EFW system can be powered long-term using AC and DC power. Two of the EFW pumps are motor-driven pumps using AC power, and the third EFW pump is turbine-driven using steam generated in the Steam Generators by decay heat. Operation of the turbine-driven pump requires only DC power. The EFW control system and flow control valves requires DC power (DC provides backup power to Instrument SUPS). In addition, the EFW flow control and isolation valves normally require instrument air which is backed up by nitrogen accumulators. The nitrogen accumulators are tested to ensure that they can provide at least 10 hours of pressure (tested via surveillance procedure STA-001-005). The EFW flow control valves can also be operated locally on the roof of the auxiliary building wing areas. In summary, assuming instrument air were not available post-fire and that the nitrogen accumulators would be discharged after 10 hours without instrument air, the EFW valves would fail open, ensuring that the EFW flow path is maintained. To prevent eventual overfilling of the Steam Generators, the operators would either replenish the accumulators or locally throttle the valves. Alternatively, they could cycle the EFW pumps (turn them off and on repetitively) to maintain the desired SG level.

EFW Suction Water Inventory

There are multiple sources of makeup available to EFW. The EFW Pumps take suction off the 170,000 gal Condensate Storage Pool (CSP). The normal makeup to the CSP is from the 500,000 gal capacity Demineralized Water Storage Tank (DWST) or the 260,000 gal capacity Condensate Storage Tank (CST), both located in the transformer yard south of the Turbine Building. Normally the DWST is aligned for makeup and a single valve, CMU-141 (Condensate Storage Pool LCV Bypass), is operated to fill the CSP. OP-003-004, Condensate Makeup, provides guidance for aligning the CST as a makeup source versus the DWST if needed.

A backup source of water for the EFW system is the Wet Cooling Tower (WCT) basins, which can be aligned to provide an additional water source to the EFW system. These basins contain

174,000 gal (WCT A) and 159,500 gal (WCT B). This use of the WCT basin water requires that manual isolation valves in the cross-connection from the WCT basins to the EFW suction lines be opened; these isolation valves are located in the hallway adjacent to the turbine-driven EFW pump. Once the isolation valves are open, the Auxiliary Component Cooling Water (ACC) pumps provide the water supply to EFW (by refilling the CSP) from the WCT basin. This action is proceduralized in OP-902-006, Loss Of Main Feedwater Recovery.

Additionally Supplemental SAMG, S-SAMG-01, provides guidance for filling the CSP directly from the Fire Protection system from two 260,000 gal capacity Fire Water Storage Tanks (FWSTs). Guidance to perform this relies on operation of one of the two available Diesel Driven Fire Pumps (or a Motor Driven Fire Pump if normal AC power is available) and then routing a fire hose to the CSP vent located in the Component Cooling Water Pump B room.

If needed, additional water could be supplied to the SGs as recommended by Severe Accident Management Guidelines (SAMG) Candidate High Level Actions. The Wet Cooling Tower (WCT) basins are open to the atmosphere and could be replenished using either (1) the onsite fire suppression system, which has 2 FWSTs with 260,000 gal of water each, 2 diesel-driven pumps, and 1 motor-driven pump; or (2) fire pumper trucks pumping from the Mississippi River. The FWSTs can be replenished from parish (county) potable water, and the Mississippi River is effectively an inexhaustible water source, so these water sources could be used indefinitely to maintain inventory supply for EFW.

Plant-specific thermal-hydraulic (T/H) calculations performed for Waterford 3 in support of post-fire safe shutdown analysis show that the EFW suction inventory needed for maintaining Mode 3 hot standby for 7 days post-fire is 769,709 gals (case using MSSVs without ADV, 383,289 gal to SG 1 and 386,420 gal to SG 2). Since this is significantly less than the available makeup inventory available from the normal methods of CSP, DWST, and CST or backup methods from WCT Basins and FWSTs, EFW can easily be maintained for significantly longer than 7 days.

Qualitative Risk Assessment:

Since there are multiple success paths for each of the functions necessary for long-term decay heat removal using EFW, the risk of failure of long-term decay heat removal is relatively small. In terms of electrical power, normal AC power is provided by offsite power; if a fire were to affect offsite power, the EDGs will provide emergency AC power. EDG fuel tanks provide over 7 days of fuel, giving ample time to plant staff to replenish the fuel tanks using readily available local fuel sources. If all AC power were to be lost, the turbine-driven EFW pump would be powered by steam (and controlled by AB battery-supplied DC power). The battery would provide DC control power for at least 4 hours (Ref. ECE91-060), giving the plant ample time to activate the Emergency Plan and staff the onsite and offsite emergency facilities; this would give the operators additional support to establish manual control of EFW or to perform other recovery actions, if necessary.

Several different means are available to control EFW flow long-term. If instrument air were lost to the EFW control and isolation valves, the nitrogen accumulators would give the operators at least 10 hours to replenish the accumulators or to assume manual control of the valves. Since the valves go open on loss of air pressure (and DC power), they would be failed in the safe position, giving the operators ample time to assume manual control.

Finally, there are multiple water sources for the EFW system. Normal available water sources include the CSP, DWST, and CST, which together can provide well over 7 days of EFW water. Additional water is available in the WCT basins, and essentially unlimited water is available from the fire suppression system with potable water makeup, or from the Mississippi River.

Because (1) each of the functions required to maintain safe and stable conditions (e.g., power, control, water inventory) has multiple success paths, (2) there is a long time available to establish alternative long-term EFW cooling configurations, and (3) the emergency facilities would be staffed and able to assist the operators, the risk of failure of long-term cooling is relatively small.

RAI PRA 06 – 90 Day Response

RAI PRA 07

Please describe how CDF and LERF are estimated in main control room (MCR) abandonment scenarios. Please describe if any fires outside of the MCR cause MCR abandonment because of loss of control and/or loss. of control room habitability and if "screening" values for post-MCR abandonment are used (e.g., CCDP of failure to successfully switch control to the primary control station (PCS) and achieve safe shutdown of 0.1), or if detailed human error analyses been completed for this activity. Please provide justification for any screening value used. If no sensitivity analysis using a CCDP higher than 0.1 was performed, the discussion cited in PRA-W3-05-006, "Waterford Fire Probabilistic Risk Assessment (FPRA) Scenarios Report," should be included.

Waterford 3 Response

The Fire PRA supporting the Waterford 3 NFPA-805 LAR assumed that a CCDP/CLERP of 0.1 was a bounding value for scenarios where the operators were required to accomplish shutdown from outside the control room. A detailed assessment was completed in response to this RAI and the assumed value (0.1) was found not to be bounding. Additional details on the results of the updated MCR abandonment analysis (including results) will be provided in response to PRA RAI 40.

The potential for a fire outside the MCR (Fire area RAB-1) that leads to MCR abandonment was evaluated. No such scenarios were found to be credible.

RAI PRA 08 – 90 Day Response

RAI PRA 09

Please describe whether the peer reviews for both the internal events PRA (IEPRA) and the FPRA consider the clarifications and qualifications from NRC Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009 (ADAMS Accession No. ML09041 0014), to the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard. If not, please provide a self-assessment of the PRA model for the RG 1.200 clarifications and qualifications and indicate how any identified gaps were dispositioned.

Waterford 3 Response

The Internal Events and Fire PRA Peer Reviews were performed by the Pressurized Water Reactor Owners Group (PWROG). The PWROG performs all PRA Peer Reviews in accordance with NEI 05-04 for Internal Events PRAs and NEI-07-12 for Fire PRAs. These NEI guidelines include reference to the RG 1.200 clarifications and qualifications which are part of the PRA peer review.

RAI PRA 10

Please identify if any VFDRs in the LAR involved performance-based evaluations of wrapped or embedded cables. If applicable, please describe how wrapped or embedded cables were modeled in the FPRA including assumptions and insights on how the PRA modeling of these cables contributes to the VFDR delta-risk evaluations.

Waterford 3 Response

Only one ERFBS (in RAB 6) is credited in the FPRA model. The existence of ERFBS (Electric Raceway Fire Barrier System – i.e. fire wrap) was not initially credited in the FPRA. The base analysis was developed assuming all cables are present in the compartment. If the resultant scenarios exhibited high risk, a refined analysis would be performed taking credit for installed ERFBS. Prior to the crediting of an ERFBS, the adequacy of the installation and fire endurance rating was verified. A discussion of the process for credit ERFBS and a listing of the ERFBS credited in the FPRA is included in PRA-W3-05-006 (Report R0247070001.06), Section 3.3. In the delta risk evaluations, all components/cables with associated VFDRs are assumed protected (set to TRUE) in the compliant case and not protected, subject to direct fire damage or random failure in the non-compliant case (whether a current fire wrap exists or not). The only exception to this is the cable in RAB6. It is treated the same in the compliant case, but is not subject to fire damage (only random failures) in the non-compliant case.

Cables within embedded conduits that do not communicate with a PAU (conduits do not penetrate into the room either at their terminus or mid-route) are not considered in the PAU (Physical Analysis Unit). No fire impact is credited for fully embedded cable. This treatment is consistent with draft NUREG-1778. The delta risk for such components would be zero as there would be no fire impact and only the associated random failures would be accounted for.

RAI PRA 11

Please identify any plant modifications (implementation items) in Attachment S of the LAR that have not been completed but which have been credited directly or indirectly in the change-in-risk estimates provided in Attachment W. When the effects of a plant modification have been included in the PRA before the modification has been completed, the models and values used in the PRA are necessarily estimates based on current plans. The as-built facility after the modification is completed may be different than the plans. Add an implementation item that, upon completion of all PRA credited implementation items, verifies the validity of the reported change-in-risk. This item should include a plan of action should the as-built change-in-risk exceed the estimates reported in the LAR.

Waterford 3 Response

Of the modifications listed in attachment S, only one S1-5 (a 3M fire wrap in RAB 2, RAB 5, RAB 6, and RAB 17) is directly credited in the PRA model. The wrap is credited in RAB 6 only. The wrapped cables in RAB 6 are assumed protected from fire in the base FPRA model. The wrap is not credited in the PRA model for the other compartments. This wrap is physically installed. None of the other modifications listed in Attachment S are credited in the FPRA model.

The delta risk values listed in Attachment W of the LAR do not credit any of the planned modifications listed in Attachment S with the exception noted above. Some additional delta risk calculations were completed to show the impact of modifications S1-3 and S1-4 (in RAB 8C and RAB 2 respectively). Table W-2 of the LAR includes notes below the table that shows the results of these additional calculations. However, the values in the body of the table and total summed value in the table do not credit the planned modifications.

The fire PRA model will be updated within six months after the completion of all the modifications listed in the Attachment S.

RAI PRA 12

Please identify any changes made to the IEPRA or FPRA since the last full-scope peer review of each of these PRA models that are consistent with the definition of a "PRA upgrade" in ASME/ANS-RA-Sa-2009, as endorsed by Regulatory Guide 1.200. Also, please address the following:

a. *If any changes are characterized as a PRA upgrade, please identify if a focused-scope peer review was performed for these changes consistent with the guidance in ASME/ANS-RA-Sa-2009, as endorsed by Regulatory Guide 1.200, and describe any findings from that focused-scope peer review and the resolution of these findings for this application.*
b. *If a focused-scope peer review has not been performed for changes characterized as a PRA upgrade, please describe what actions will be implemented to address this review deficiency.*

Waterford 3 Response

Calculation PRA-W3-05-003 defines the changes to the FPIE model to support the Fire PRA. The majority of these changes are related to adding events to support MSO and FPRA credited equipment. These changes were reviewed by the Fire PRA Peer Review team and are not PRA upgrades.

In addition, Waterford has made 8 changes documented in Model Change Requests primarily to address changes to Waterford EOOS monitor to facilitate rolling out the Revision 4 model update and address operator issues discovered during EOOS use. None of these EOOS changes are changes to the methodology reviewed by the PRA Peer Review Team and are not PRA Upgrades.

RAI PRA 13

With respect to NUREG-1921, "EPRIINRC-RES Fire Human Reliability Analysis Guidelines," dated May 2012 (ADAMS Accession No. ML 12167A070), please describe how the screening values assumed for human failure events (HFEs) in the fire HRA (see PRA-W3-05-00

5-3 through 5-7, §5.2.1, Single HFE, §5.2.2, Joint HFE Screening Values) compare against those developed in NUREG-1921. Please describe if application of the latter rather than the former would have retained any HFEs that were screened out.

Also, with respect to PRA-W3-05-003, pages D-3 through D-7, Table D-2, Human Error Probability (HEP) Calculation Input Considerations for FPRA, please describe to what extent was the guidance in Appendix C of NUREG-1921 considered for any adjustments due to fire. If Appendix C of NUREG-1921 was not considered, please describe how the adjustments made were different. Further, during the audit, reference was made to use of an EPRI "HRA Toolbox" in lieu of the "HRA Calculator" -please explain what is the HRA Toolbox and how is it different from the HRA calculator.

In addition, please describe if the criteria provided in NUREG-1921, §5.1.3 and §4.3, were considered when assuming a bounding CCDP = 0.1 for MCR abandonment (see

PRA-W3-05-006, pages 13-4, and 13-6 through 13-20, § 13.2.1, MCR Abandonment Time). If not, please address the potential effect of such consideration.

Waterford 3 Response

Waterford 3 Fire PRA HRA vs. NUREG-1921

In the Waterford 3 (W3) Fire PRA, no Human Failure Events (HFEs) were screened out. “Screening” values for Human Error Probabilities (HEPs) are conservative values for the HEPs calculated as described in “PRA-W3-05-003, “Waterford FPRA Quantification Model Preparation and Database Development Report.” These conservative “screening” HEPs were entered into the Fire PRA (FPRA) model for the initial quantification and retained in the model. After the initial quantifications, any HFEs that were significant contributors to the fire risk results were subjected to more detailed HRA in order to estimate more realistic HEPs for the particular fire scenarios. But all the HFEs are included in the final FPRA results, with either these initial conservative screening values or with the more realistic HEPs from the detailed fire HRA analysis. (Waterford 3 FPRA Summary Report, PRA-W3-05-007, documents that the NUREG/CR-6850 Quantitative Screening, Task 7, was not used for Waterford 3.)

Waterford 3 Fire PRA HRA Calculation Input vs. NUREG-1921 Appendix C

Appendix C of NURG-1921 describes the quantification of fire HEPs using the EPRI HRA methods as implemented in the HRA Calculator. These methods are documented in two EPRI reports: (1) EPRI TR-100259, *An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment,* 1992; and (2) EPRI TR-6937, *Operator Reliability Experiments Using Nuclear Power Plant Simulators*, July 1990. The Waterford 3 FPIE PRA and Fire PRA both used these HRA methods. While the HRA Calculator was not used, an earlier, proprietary implementation of the EPRI methods, called the “HRA Toolbox”, was used.

The use of the EPRI HRA methods in the Waterford 3 FPIE PRA is described in detail in calculation PRA-W3-01-001S03, “Waterford 3 PSA At-Power Level 1 Human Reliability Analysis”. The EPRI HCR/ORE and Cause Based Decision Tree Method (CBDTM) methods for cognitive errors and THERP for execution errors, just as described in NUREG-1921, Appendix C. The performance shaping factors in NUREG-1921 (e.g., Table C-3) are the same as used in the Waterford 3 FPIE PRA. In performing the detailed HRA for the Waterford 3 Fire PRA, the same HRA method was used. In conclusion, with respect to PRA-W3-05-003, pp. D-3 thru D-7,

Table D-2, HEP Calculation Input Considerations for FPRA, the guidance in App. C of NUREG-1921 was followed in making adjustments due to fire. In addition, the “HRA Toolbox” is equivalent to the HRA Calculator since both HRA software systems use the same methods and data.

Use of CCDP = 0.1 for MCR Abandonment

Waterford is currently working on a revised MCR abandonment evaluation. This updated evaluation includes a more detailed human reliability evaluation and does not use an assumed value for CCDP. Additional details of the MCR abandonment will be provided in the response to RAI PRA 40.

RAI PRA 14

Please describe how changes to the seismicities as a result of the United States Geological Survey (USGS) re-evaluation for the central and eastern U.S. (USGS, "2008 NSHM Gridded Data, Peak Ground Acceleration") were considered in PRA-W3-05-004, "Waterford FPRA Seismic/Fire Interaction Report." In particular, please discuss the following (confirm if considered; if not, provide, at least, a qualitative disposition):

a. *Please discuss the applicability of the seismic-fire interaction analysis performed for the Individual Plant Examination of External Events (IPEEE) to the current state of seismic-fire interactions, with respect to the discussion on pages 2-2 and 2-4, §2.1, Seismically-Induced Fires; and page 2-5, §2-2, Degradation of Fire Suppression Systems and Features, which states "[T]he S[eismic] R[eview) T[eam] screened generic classes of equipment consistent with the guidance given in Appendix D of EPRI NP 6041-SL ... EPRI NP-6041-SL ... was used as the basis for developing the list of equipment used in the seismic margins assessment ... There was a 100 percent walkdown of safe shutdown (SSD) equipment for seismic interactions [t]o verify the seismic adequacy ... using the EPRI NP-6041 SL methodology ...”*
b. *Potential changes to the frequencies per acceleration level in Table A-1, which are based on the Lawrence Livermore National Laboratory (LLNL) seismic hazard estimates from 1994 (NUREG-1488, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," April 1994 (ADAMS Accession No. ML052640591», as presented on page Att. 1-2, Attachment 1, Table 1 -Seismic Hazard. (For example, describe if the conclusion on page Att. 1-3 regarding the initiation frequency for a large spill [and fire] from the diesel generator (DG) oil and fuel oil tanks would be affected.)*

Waterford 3 Response

The intent of the Seismic-Fire Interaction from the ASME/ANS Standard is:

> “The Fire PRA shall include a qualitative assessment of potential seismic/fire interaction issues in the Fire PRA.”

The Waterford Seismic-Fire interaction report is a qualitative assessment to look primarily at vulnerabilities due to structural soundness, equipment anchorage, impacts on fire brigade and automatic suppression. No vulnerabilities exist. Since this evaluation does not (and is not required to) evaluate impacts on seismic frequencies, the USGS report does not impact this evaluation.

a. Two areas were generically screened as part of the Waterford Reduced Scope Seismic Margins analysis. The structures at Waterford 3 were screened generically. The drawings and analysis models were reviewed for details that might indicate seismic vulnerabilities and confirmed that consistent good practice in design detail and analysis was implemented at Waterford 3. A structural walkdown was performed to review a sample of the details of connections, reinforcement bar placement, construction joints, etc. Distribution systems that were installed in bulk such as piping, cable trays, HVAC ducting, electrical conduit, and instrument lines were screened generically after completion of a walkdown with verification that the distribution systems meet the inclusion rules. The Seismic Margins walkdown included all components from the SSEL

b. The Seismic Fire Interaction report (page Attachment 1-3) states that previous seismic studies have indicated that large tanks such as the storage tank for fuel oil should exhibit a capacity in excess of 1.0g. Based on the USGS Seismic Hazard Curves, the frequency of a seismic event in excess of 1.0g is 2.99E-8/yr. This frequency is less than the frequency of 9.31E-8/yr shown in the Seismic-Fire Interaction report. Therefore, the conclusion regarding the large oil tank spills is not impacted by the latest USGS report.

<u>RAI PRA 15 – 90 Day Response</u>

<u>RAI PRA 16 – 90 Day Response</u>

<u>RAI PRA 17</u>

In the Transition Report, on pages V-3 through V-16, Attachment V, Table V-1, FPRA Peer Review Facts and Observations (F&Os), F&O FSS-C1-01 states that 'These results are comparable to the results of more detailed fire modeling." F&Os FSS-H2-01 and FSS-H5-01 state that " ... [N]o detailed fire modeling was done ... The Waterford 3 FPRA uses generic fire modeling for individual scenarios." Please describe the basis for these conclusions.

Typically, "generic" fire modeling is used for screening to conservatively identify potential targets and ignition sources based on conservative assumptions (e.g., ZOIs for 98th percentile HRR fires). (See also W3F1-2012-0005, "Supplemental Information in Support of Waterford 3 LAR Acceptance Review," page 1 of 2, Att. 2, Table V-2, Fire PRA -Category I Summary, with respect to supporting requirements (SRs) FSS-C1, -C2, -C3, -E3 and -H2.) In addition, the general discussion on pages 0-3 and 0-4 of §0.1 in PRA-W3-05-007, "Waterford FPRA Summary Report," cites use of generic fire modeling treatments as a means of reducing conservatism in lieu of conservative scoping analysis techniques, and justification for not performing detailed fire modeling other than for MCR abandonment. In fact, the generic treatment may be non-conservative and not necessarily realistic, such that employing it during the screening phase could lead to omission of potentially risk-significant scenarios. Coupled with other non-conservative assumptions, such as no fire spread among thermoset cables, it is likely that considerable uncertainty has been introduced in Tasks 8 and 11, not necessarily offset by the alleged conservatism in the HRRs from NUREG/CR-6850. Also, for Task 10, additional non-conservatism may have been introduced by the inadvertent crediting of CPTs as a means of reducing spurious action probability, to be addressed via Sensitivity Analysis (see PRA RAI42).

Please describe for other than for MCR abandonment where detailed fire modeling is performed consistent with Capability Category II. If none, please discuss how the bounding analyses at the Capability Category I level assures the risk and delta-risk results are representative.

Waterford 3 Response

The modeled heat release rates (HRR) for analyzed potential fires were addressed in a generic manner for the baseline assessment and utilized conservative HRRs with the exception of transient fires.

The use of an unapproved method (UAM) in the baseline study did somewhat limit the transient fire HRR for the baseline analysis and a sensitivity analysis of the transient cases using a conservative value (98th percentile) has been performed subsequent to the baseline analysis to support the response to PRA 2, PRA 3, PRA 18 and PRA 20.

The results from these analyses indicate that there is a relative measurable increase in core damage frequency (CDF) and large early release frequency (LERF) for the transient cases but that overall the increase is not significant with regard to conclusions drawn by the baseline assessment and the selection of important accident scenarios in terms of overall CDF or LERF.

The fire modeling has been updated to address this assumed HRR and the zone of influence (ZOI) adjusted as appropriate. In addition, the sensitive component ZOI has been expanded to meet the suggested criterion in NUREG/CR-6850 (see responses to RAIs PRA 16 and PRA 20) and reevaluated for the transient cases. The results support an increase in potential component impacts, but it does not result in a change in overall conclusions or accident sequences when using the 98th percentile HRR for transient fires (317 kW).

No other detailed fire modeling activities (such as plant-specific Fire Dynamics Simulation (FDS) analyses) were performed as part of the internal fire PRA. However, other prior studies have been performed for Waterford 3 using plant-specific information and detailed FDS models as part of the plant's Appendix R program. A comparison was made between the predicted temperature impacts based on the methodology for the generic predictions (which uses the NUREG-1805 Fire Dynamics Tools or FDT spreadsheets) and these prior FDS analyses for the significant PAUs (RAB 7A, 7B, 8A, and 8B) to determine how well the generic assessment enveloped the more specific assessment.

In the detailed FDS work, for a transient scenario with an overall HRR of 636 kW (a 159 kW source located in a corner that receives a multiplication factor of 4 as described in the generic fire modeling report) the FDS predicted ZOI was that a component or cable with a 0.2 m radial offset and 2 m elevation above the fire source would be outside of the zone of influence and have no impacts (not be failed) from the fire scenario and that directly above fire source at 1.98 m elevation above the fire source was failed (however, the detailed FDS work but does not give the distance directly above fire source when the criteria would not be failed).

The FDT plume temperature (the same methods as the generic fire modeling) results for the same scenario setup predicts a ZOI of 3 m vertical (for the same failure criteria as listed in the detailed FDS report); noting that radial offset temperature predictions cannot be done within a FDT evaluation. This demonstrates an example of the generic approach methods predicting more restrictive ZOIs which would consequentially predict more component or cable failures for a given scenario.

RAI PRA 18 – 90 Day Response

RAI PRA 19

Concluding that ignition frequency is relatively insensitive to the results of ignition source counting assumes (1) there are a lot of ignition sources of a particular bin type throughout the plant (denominator effect) and (2) there are a substantial number of similar sources in the particular PAU of concern (numerator effect). (See PRA-W3-05-007, page 2-8, §2-15, Uncertainty and Sensitivity Analysis.) If either is low, the results can be fairly sensitive to small changes.

Please describe whether the accuracy of counts for ignition source bins with either a low total number throughout the plant or within a particular PAU was checked. With regard to SR IGN-A-10 and B5, on page 0-6, §D.2, Discussion Related to Specific Standard Requirements, ignition frequencies are cited as conservative because they arise from NUREG/CR-6850. Whether the original or FAQ values are used, probability distributions are provided, such that at least the parametric uncertainty should be addressed.

Waterford 3 Response

The original ignition source count was performed using a preparer and verifier as documented in Report PRA-W3-05-001. Subsequently, during scenario development, the ignition sources were again verified by the scenario preparers. All sources were specifically observed as part of the scenario development and it was determined whether a scenario was required for the specific sources (i.e., there could be a source target interaction based on maximum ZOI). This provided a redundant verification of the counting. This process is very sensitive to low counts either in the numerator or denominator given the specific activity of trying to locate the smaller number of items.

No quantitative uncertainty was completed associated with ignition frequency development. A qualitative discussion on ignition frequency uncertainty is provided in Appendix D of PRA-W3-05-007. The peer review conducted for the Waterford FPRA effort noted IGN-A10 as being met at a Capability Category 2 level (though a finding was assigned) and noted that all listed requirements associated with IGN-B5 were met.

RAI PRA 20 – 90 Day Response

RAI PRA 21

When selecting "targets," please describe whether the possibility of a "target" beyond the nominal ZOI of an ignition source, but within range of damage if the fire propagates beyond the ZOI, was included (e.g., a "target" cable tray, beyond the ZOI, but near enough to another that is within the ZOI such that fire spread to that tray could subsequently damage the "target" if the fire goes unsuppressed). (See PRA-W3-05-003, page 2-2, §2.0, Methodology Review.) Also, please describe how the potential for fire spread once it escapes the panel and cables in contact with the panel was addressed. (See PRA-W3-05-006, pages 2-8 and 2-9, §2.2, Assumption 10.)

Waterford 3 Response

For the IEEE-383 qualified thermoset cables at Waterford, the fire spread is considered to be limited to those raceways within the cable damage ZOI. Spread of a fire beyond the ZOI due to a secondary (i.e. target) fire was not evaluated. Damage was limited by the initial source ZOI.

How fire spread was treated for cables in contact with a panel depended on the panel. For well sealed panels, assumption 10 (PRA-W3-05-006) explains the treatment. As noted in assumption 10, cables in contact with the panel were included as targets for the well-sealed panels. The configuration of the well-sealed panels precludes fire spread beyond the panel given the limited ventilation precluding a significant fire which would damage targets other those associated with cables potentially in contact with the panel.

Fires associated with non-well-sealed cabinets impact all cables/targets in the associated ZOI (not just ones in contact with the source).

RAI PRA 22

Please describe why the Appendix L methods from NUREG/CR-6850 are not appropriate for at least fires within a single cabinet. Appendix L provides an approach for estimating the probability of unsuppressed damage in the main control board (MCB) based on distance between assumed "targets," with distances small enough to lie well within a single cabinet. The approach, while based on an assumed MCB cumulative panel width of 60 ft, is intended to apply to all but the narrowest MCB panels as well, since the effect of developing the approach using a narrower panel width would have been small. Credit also appears to be taken for "segregated" wiring within selective panels as the basis for assuming only trains of single systems would be impacted by an MCB fire. The basis is cabinet fire growth rates (presumably the 12-min average estimated in NUREG/CR-68S0) and flame spread rate for IEEE-383 qualified cables/wires. Note that 12 min is an average growth time, and shorter ones are possible, likely where wires, instead of cables, predominate as the combustible. Furthermore, flame spread rate, which is based on cable type (thermoset vs. thermoplastic), not IEEE-383 qualification, may be much higher for wires than for cables. The technique of Appendix L is intended to take such factors into account, serving as a surrogate for the inability to take credit such as is postulated (See PRA-W3-0S-006, pages 13-1 and 13-2, §13.1, MCR Analysis.). Please provide a comparison between the approach to modeling fire damage within a single MCB panel vs. that of Appendix L if the former is deemed bounding. (See also PRA-W3-05-007, page C-4, §C.2.4, RAB1A-E015.)

Waterford 3 Response

Upon evaluation and review, Waterford concludes that NUREG/CR-6850 Appendix L methods are appropriate and should be used. Ongoing efforts to utilize Appendix L methods are underway. The impact of the revised approach based on the NUREG/CR 6850 modeling will be provided in the response to RAI PRA 40.

RAI PRA 23

While potential conservatisms present in the Hughes Approach are credited as a basis for not performing detailed fire modeling, the presence of non-conservative aspects, such as the assumption of only 69 kW HRRs for level 2 transients or thermoset damage thresholds for

sensitive electronics, may render this conclusion questionable. The reliance on information available from the fire events database (FEDB), complete only through 2000 and currently being updated, as justification for performing no detailed fire modeling does not recognize that such data may be incomplete and, therefore, non-conservatively adapted outside the consensus approach of NUREG/CR-6850. (See PRA-W3-05-006, pages 14-1 and 14-2, §14.0, Use of Generic Fire Modeling Treatments vs. Detailed Fire Modeling.) Please provide additional justification for not performing detailed fire modeling.

Waterford 3 Response

The modeled heat release rates (HRR) for analyzed potential fires were addressed in a generic manner for the baseline assessment and utilized conservative HRRs with the exception of transient fires. The use of a UAM did somewhat limit the transient fire HRR for the baseline analysis and a sensitivity analysis of the transient cases using a conservative value (98th percentile) has been performed subsequent to the baseline analysis. This sensitivity analysis does indicate that there is a relative measurable increase for the transient cases but that overall the increase is not significant with regard to conclusions drawn by the baseline assessment and the selection of important accident scenarios. The fire modeling has been updated to address this assumed HRR and the zone of influence (ZOI) adjusted as appropriate. In addition, the sensitive component ZOI has been expanded to meet the suggested criterion in NUREG/CR-6850 (see responses to RAIs PRA 16 and PRA 20) and reevaluated for the transient cases. The results again support an increase in potential component impacts, but it does not result in a change in overall conclusions or accident sequences when using the 98th percentile HRR for transient fires (317 kW).

No other detailed fire modeling activities (such as plant-specific PAU or multi-compartment FDS analyses) were performed as part of the internal fire PRA. However, other prior detailed studies have been performed at Waterford 3 using plant-specific information and detailed FDS models as part of the plant's Appendix R program. It would generally be expected that the more dominant PAUs to the internal fire PRA model results would receive additional, more detailed and refined fire modeling treatments. It was stated in the Waterford internal fire PRA report PRA-W3-05-006 that such activities were not undertaken to enable easier configuration control and updates of the fire PRA.

A comparison was made between the predicted temperature impacts based on the methodology for the generic predictions (which uses the NUREG-1805 Fire Dynamic Tools or FDT spreadsheets) and these prior detailed FDS analyses for the significant PAUs (RAB 7A, 7B, 8A, and 8B) to determine how well the generic assessment enveloped the more specific assessment. Further discussion and comparison of the generic fire modeling and detailed fire modeling are presented in the response to PRA RAI 17.

The Waterford internal fire PRA model results, using the currently defined methods of the study, produced results that were not indicative of the need for more detailed fire modeling treatments. However, with several methods containing UAMs, should the quantitative impacts of the other sensitivity studies for RAI responses indicate more restrictive outcomes of plant PAUs for CDF or LERF results, more detailed and plant specific fire modeling could be used to more realistically estimate those PAU scenario impacts and to refine the internal fire PRA model. The quantitative impacts will be presented in the response to PRA RAI 40.

RAI PRA 24

In the Transition Report, page W-3, Attachment W, §W.1, Consideration for External Events, it is stated that "Waterford 3 has no high winds, floods, or off-site industrial facility accidents that significantly alter the Waterford 3 estimate of either CDF or the distribution of containment release categories." Given the potential for hurricanes (more severe than Katrina) and tornadoes, provide the basis for dismissal of high wind risk. Also, in W3F1-2012-0005, page 6 of 7, SQ 4, Site Risk from External Events, the estimate for seismic CDF presumes an "effective" CCDP of ~, 0.07 (via [1 E-6]/[1.SE-5]), including fragilities, such that CDF ~ (1.SE-5/yr)(0.07) ~ 1 E-6/yr (with LERF a factor of 10 lower). Based on the IPEEE Seismic Margins Analysis (review level earthquake of 0.1g), the CDF for seismic using the updated USGS hazard curves is estimated to be about 1.8E-5/year. Please explain the basis for your lower estimate of the seismic CDF.

Waterford 3 Response

High Wind Events:

The magnitude of high winds is bounded by the values in UFSAR Chapter 3.3. The plant structures defined as seismic Category I structures are designed for a maximum sustained wind of 200 mph at 30 feet above plant grade. Those same seismic Category I structures were designed to resist a tornado of 300 mph tangential wind velocity and a 60 mph translational wind velocity.

The design frequency of hurricanes in the IPEEE is calculated as 0.3/year based on the data in UFSAR section 2.3.1.2.2. The tornado frequency from UFSAR Section 2.3 is one per twelve months. These frequencies are still considered valid for Waterford.

Seismic CDF Estimate:

NRC seismic hazard estimates in the GI-199 Safety / Risk Assessment used an extremely conservative method which assumed that the Waterford-3 plant was represented by a HCLPF (High Confidence of Low Probability of Failure) corresponding to the Safe Shutdown Earthquake (SSE) value of 0.1g. This results in a significant underestimate of the HCLPF. Because of the low seismic acceleration for the Waterford-3 SSE, the concrete and steel structures, designed for seismic, tornado and other severe loads, will have a failure capacity much higher than the SSE. The anchorages for equipment, which the walkdown found rugged, will similarly have margin.

Since the seismic hazard estimate for Waterford in GI-199 is so conservative, a slightly more realistic seismic CDF is calculated using the Waterford hazard curves. This estimate assumes that at the SSE value, both safety-related buses should be available to mitigate the event. From NRC Inspection Manual 609, Appendix A, Table 5, the failure probability of a multi-train system is approximately 1E-3. This probability is increased to 1E-2 to account for any increase failure probabilities based on the seismic event. If the seismic event is double the SSE amplitude, one safety-related train is expected to be available to mitigate the risk. From NRC Inspection Manual 609, the failure probability of a single train is approximately 1E-2. Again this probability is increased by an order of magnitude to account for the seismic event. At 3 times the SSE amplitude, the CCDP is estimated to be 0.5 based on some mitigating equipment available requiring an operator action to mitigate core damage. Finally, if the seismic event is 5 times the SSE amplitude, it is assumed that all mitigation equipment is failed and the CCDP is 1.0. Based on the walkdowns performed for the IPEEE, these failure probabilities are considered to be

conservative. The LERF frequency is estimated to be at least an order of magnitude below the CDF based on the results of the internal events and fire PRA models.

Amplitude (g)	Mean Frequency (/yr)	Estimated CCDP	Seismic CDF (/yr)
0.1	1.53E-5	0.01	1.53E-7/yr
0.2	1.95E-6	0.1	1.95E-7/yr
0.3	7.17E-7	0.5	3.59E-7/yr
0.5	1.95E-7	1	1.95E-7/yr

Based on these conservative estimates above, the seismic CDF for Waterford is 9.02E-7/yr. Since LERF is taken to be an order of magnitude less than CDF, the Waterford seismic LERF is 9E-8/yr

RAI PRA 25 – 90 Day Response

RAI PRA 26 – 90 Day Response

RAI PRA 27 – 90 Day Response

RAI PRA 28

In PRA-W3-05-003, on page B-2, Appendix B, Method, while the basis for assuming 1 E-6 for the probability floor on joint HEPs (JHEPs) within a single cutset is provided, describe what would be the effect of using the NUREG-1792 suggested value of 1E-5. Please describe whether any combinations set to the minimum of I E-6 become potentially risk significant if set to a minimum of 1 E-5.

Waterford 3 Response

In an effort to reflect the relative increase in CDF if a higher Joint HEP floor value were used, the Joint HEP values that were less than 1E-5 were increased to 1E-5 within the aggregate cutset file. The Fire CDF obtained with a JHEP floor value of 1E-6 was 3.431E-5. When the JHEPs were increased to a minimum of 1E-5, the CDF increased to 3.554E-5 (3.5%). The non-risk-significant dependent actions remained non-risk significant when increased to 1E-5.

				JHEP=1E-5		JHEP<1E-5	
Event Name	Probability	Fus Ves	BirnBm	Red W	Ach W	Red W	Ach W
ZHF-C2-002	1.00E-6	5.53E-5	1.90E-3	1.0005	54.3	1.0001	56.2
ZHF-C2-016	1.00E-6	3.32E-5	1.14E-3	1.0003	33.1	1.0000	34.2
ZHF-C2-019	1.00E-6	3.68E-5	1.26E-3	1.0004	36.5	1.0000	37.8

				JHEP=1E-5		JHEP<1E-5	
Event Name	Probability	Fus Ves	BirnBm	Red W	Ach W	Red W	Ach W
ZHF-C2-028	1.00E-6	2.60E-9	8.93E-8	1.0000	1.00	1.0000	1.00
ZHF-C2-031	1.00E-6	5.53E-5	1.90E-3	1.0005	54.3	1.0001	56.2
ZHF-C3-001	1.00E-6	3.22E-7	1.10E-5	1.0000	1.31	1.0000	1.32
ZHF-C3-002	1.00E-6	1.95E-3	6.48E-2	1.0192	1820	1.0020	1890
ZHF-C3-003	1.00E-6	1.34E-5	4.60E-4	1.0001	14.0	1.0000	14.4
ZHF-C3-004	4.30E-6	1.27E-8	1.02E-7	1.0000	1.00	1.0000	1.00
ZHF-C3-005	1.00E-6	1.13E-5	3.89E-4	1.0001	11.9	1.0000	12.3
ZHF-C3-007	1.00E-6	1.83E-3	6.07E-2	1.0179	1710	1.0018	1770
ZHF-C3-013	2.50E-6	7.90E-6	1.08E-4	1.0000	4.05	1.0000	4.16
ZHF-C3-018	4.90E-6	1.39E-10	9.77E-10	1.0000	1.00	1.0000	1.00

Note - The aggregate cutset file used to develop this response is one from quantification completed to support the LAR submittal (late 2011). While ongoing work may impact the model and resulting cutsets, the finding of the above sensitivity (that a higher joint HEP value has small impact on risk) will likely be unchanged.

RAI PRA 29

For PRA-W3-05-005, page 1-1, §1.0, Scope, please describe how that methodology differs from that of NUREG/CR-6850 but still satisfies ASME/ANS RA-Sb-2009.

Waterford 3 Response

The main difference in the screening methodology is the cornerstone to the method which is based on a correlation of the potential for a hot gas layer (HGL) to form and the physical analysis unit (PAU) volume. The larger the size of the volume the lower predicted peak temperatures and less chance for a hot gas layer to form. Rather than determining what the required heat release rate (HRR) would be for a PAU to form a damaging HGL, the methodology defines the minimum PAU internal volume that is sufficient to preclude the formation of a damaging HGL for the specified PAU scenario HRR. The method uses assumed PAU ceiling heights of 7 or 12 feet depending on which heat release rate (HRR) group is being evaluated, and the PAU floor area with consideration for equipment occupancy removed from the overall free air volume calculation. The use of such heights may be accurate to plant conditions, but it is not stated that all analyzed PAUs have a ceiling height of 12 feet for the transient scenarios (based on a fire source at floor level) nor is it stated that all electrical panels or cabinets are of a certain height to base the 7 foot height for the panel scenarios on as ceiling height minus panel top height.

An additional difference is that an ignition frequency correction factor (IGFCF) was calculated to provide partitioning factors for those scenarios with damage beyond the ignition source. This factor is based on information contained in the Supplemental PRA Methods, Table 2-2.

Qualitative screenings are not undertaken in the current methodology. As a first step, PAUs with infinite volume (such as PAUs located outside with unlimited air space) or with fixed volumes that are greater than 14,000 cubic feet are screened based on the methodology that such a PAU volume will not allow a HGL to form given an electrical panel fire with secondary ignition of combustibles within the source fire's zone of influence.

Zones that are retained from this step are evaluated for either 69 kW or 702 kW HRR groupings depending on what the particular scenario and the PAU scenarios are. In addition, there are scenarios for both the 69 kW and 702 kW groupings that include secondary ignitions with overall HRRs of 1000 kW and 1750 kW for the 69 kW and 702 kW groupings respectively.

Screening of the open or unlimited free air volume PAUs is considered appropriate, though more justification or presentation of analysis results to justify the use of the 14,000 cubic foot screening size may be useful for future updates, but it is not inconsistent with ASME/ANS RA-Sb-2009 guidance.

Further frequency based screenings (based on the probability of a HGL forming) are undertaken that account for the source's ignition frequency, non-suppression probabilities for manual and automatic suppression, severity factor for the particular scenario, and for the case of the 702 kW HRR grouping the ignition frequency correction factor and the probability of barrier failure for zones enclosed with fire rated boundaries only.

The NUREG/CR-6850 methodology for multi-compartment scenarios allows for qualitative screenings, for screenings of PAUs that do not contain ignition sources with sufficient HRRs to produce a HGL, for screenings based on the probability of a hot gas layer forming based on ignition frequency, non-suppression factors, severity factors, and barrier failure probabilities, as well as screenings for low CDF contributions of scenarios.

ASME/ANS RA-Sb-2009 requires that a screening methodology be applied. The current analysis defines a methodology and applies that methodology as described above. However, using the actual PAU height along with the PAU's floor area to calculate and present the actual estimated PAU HGL temperature for the expected scenario HRR with a comparison to the critical failure temperature for the PAU targets (cables, components and potentially sensitive electronics) while not accounting for any ignition frequency correction factors would be a more directly comparable methodology to the NUREG/CR-6850 proposed analysis. This would allow for an easier interpretation of the methodology, its use, and predicted results.

For the two limits defined in the baseline assessment, the 69 kW may have initially limited the selection for HGL screening for transient selections. However the revised transient assessment utilizing 317 kW HRR in support of responses to RAIs PRA 02 and PRA 18 addresses potential for HGL development and this would be expected to be bounded by the results for 702 kW screening.

<u>RAI PRA 30</u>

In PRA-W3-05-005, page 1-5, Appendix I, Manual Suppression Term, the apparently generic non-suppression (NSP) factors used for the screening are not necessarily bounding, as they apply only to electrical fires. Several other types of fires yield higher NSPs, including what one might expect as more appropriate as a generic factor, being that for "all fires" from NUREG/CR-6850, Supplement 1 (FAQ 08-0050). Also, the NSP for 60 min for an electrical fire

should be 0.002, not 0.001, which may seem insignificant except that NSP is a multiplicative term, thereby potentially yielding a factor of -2 underestimate during screening. Please provide the basis for the NSP curves selected as bounding and a sensitivity analysis where the minimum NSP is reduced to 0.001.

Waterford 3 Response

The suppression term used in PRA-W3-05-005 is limited to response for a fire caused by electrical reasons as noted. Given the UAM approach for transient fires and oil fires the selection may not have been the most conservative, but did represent the curve for the most predominant source which was cabinet fires. The curves specifically state the fire source and the table for electrical fires is not bounding as noted although the differences for specific values are small. The “All Fires” column would be a more appropriate selection.

The UAM will be addressed as a part of the sensitivity cases performed in response to RAI PRA 40 and the impact request for RAI PRA 02. These cases have incorporated the appropriate manual suppression terms and will provide the basis for comparison as to the sensitivity of the results on this baseline assumption.

RAI PRA 31

PRA-W3-05-006, page 2-5, §2.1, Industry Standards, Table 2-1 cites disposition of FAQ 08-0051 as "incorporated" although its status is cited as "under review." Since this FAQ has now been closed, and incorporated into Supplement 1 to NUREG/CR-6850, please describe whether this disposition is consistent with the final version of the FAQ.

Waterford 3 Response

The final FAQ disposition has been considered in the Waterford FPRA. The ‘Status’ column in Table 2-1 is in error. The note at the bottom of Table 2-1 (“The status refers to the review and acceptance by the Pilot effort and NRC at the time this notebook was issued”) was accurate when the document was drafted, but the FAQ was closed prior to document finalization and submittal of the LAR. The closeout memo for the FAQ is referenced in the LAR (Section H-1 of the LAR).

RAI PRA 32 – 90 Day Response

RAI PRA 33

PRA-W3-05-006, page 2-9, §2.2, Assumptions, for Assumption 10, please describe how the potential for fire spread once it escapes the panel and cables in contact with the panel is addressed. For Assumption 11, please describe whether the location of the fire for objects placed on the 'floor was elevated sufficiently to justify the assumed height for the transient fire source (e.g., a trash can 'fire). (See also, page 6-2, §6.2, Transient Locations.) If not, please describe why not. If yes, provide examples.

Waterford 3 Response

The potential for fire propagation through cables was not explicitly considered within the baseline assessment on the assumption that the failure of components within the fire zone of influence will bound the impact of fire propagation.

Recent walkdown activities have confirmed that transient fire locations are properly elevated and locations are documented in the transient assessments based on the walkdown findings.

RAI PRA 34

PRA-W3-05-006, page 10-3, §10.3, Potential for Structural Collapse, FAQ 08-0044, not FAQ 08-0050, should be the reference for the split fraction for a large oil spill fire from an MFWP. If large (10 percent or more) is sufficient for damage, the split fraction should be 0.034, not 0.02. If very large (100 percent) is required, the split fraction should be 0.0034. The CDF estimate should be recalculated if the former is correct, although it would not more than double. Also, please describe the basis for assuming the CCDP =0.01 is presumably bounding.

Waterford 3 Response

A sensitivity case was conducted for the potential for structural collapse based on the split fractions from FAQ 08-0044. As a part of the sensitivity analysis and in response to the RAI, a review of the assumption that the CCDP of 0.01 is bounding was conducted and it was determined to be non-conservative. The impacts of revising this assumption and including the structural collapse of the TGB as a specific scenario in the WSES 3 Fire PRA will be included in the cumulative quantitative impacts evaluated in response to RAI PRA 40.

RAI PRA 35

In PRA-W3-05-006, page 11-1, §11.0, Advanced Fire PRA Methods, reference is made to the "means to determine the maximum heat release rate for a ventilated electrical cabinet [in Appendix F]." However, there is no discussion of this in Appendix F. Please clarify.

Waterford 3 Response

Cabinet heat release rates (or the means to determine them) are not provided in F. Table B-1 (pages B-6 and B-7 of PRA-W3-05-006) shows the heat release rate ranges used in the scoping fire modeling. The table provides scoping fire modeling separation distance data. The HRR ranges for both open and vented cabinets, for both 383 and non-383 cable are listed in the table.

Section 3.2.1 of Attachment 1 of PRA-W3-05-006 discusses the method used to develop heat release rates for electrical cabinets in the MCR abandonment time evaluation.

RAI PRA 36 – 90 Day Response

RAI PRA 37 – 90 Day Response

RAI PRA 38

In PRA-W3-05-007, page D-2, §D.1, General Discussion, conservatism in the ignition frequency data for Task 6 is cited, as well as its link to conservatism in non-suppression probability data. Given both of these have been re-evaluated via FAQs (08-0048 and 08-0050), with the frequency values generally reduced by a factor of -2 and the NSPs allowed to credit non-brigade suppression response, citing of the combined conservatism as significant'" seems inappropriate and may mask the fact that uncertainty, not conservatism, is the more important concern. Also, on page D-6, §D.2, Discussion Related to Specific Standard Requirements, there is a similar citation on conservatism with regard to SR CF-A2. If the crediting for CPTs is extensive, there may be non-conservatism, rather than excessive conservatism present. Please provide clarification for these issues.

Waterford 3 Response

In consideration of the re-evaluation of ignition frequency and non-suppression probability data via FAQs 08-0048 and 08-0050, it is more appropriate to describe the ignition frequency data as possessing “some” conservatism, rather than “significant” conservatism. This applies also to the combined effect of ignition frequency data and non-suppression probabilities. It is believed that the heat release rates associated with the ignition frequency bins are generally conservative, but cannot go so far as to conclude that they are significantly conservative. However, the ignition frequency data, and associated heat release rates, are characterized by uncertainty primarily biased in the conservative direction.

PRA-W3-05-006, “WSES Fire PRA Fire Scenarios Report,” Appendix C, Cable Failure Mode and Likelihood Initial Assessment, lists the components for which cable failure probabilities were calculated using the methods in NUREG/CR-6850, Task 10. For those circuits which included a Control Power Transformer (CPT), the NUREG/CR-6850 failure probabilities for circuits including a CPT were used. Less than 22% of the cables in the Waterford 3 FPRA credit CPTs. Therefore, the impact of potential changes in the NUREG/CR 6850 specified circuit failure probability for hot shorts for circuits with CPTs is not expected to be significant.

RAI PRA 39

In PRA-W3-05-007, Appendix F, Importance Measure Results, based on the dominant fire scenario contributors to CDF, one might expect to see the following components included among basic events whose importances (F-V) are fairly high:

a. *SSD ESWGR 31AB*
b. *Panel Aux Panel 4A through 40*
c. *SWGR 7kV ESWGR 1B*
d. *Control Panel CP-10*
e. *4kV ESWGR 3B*
f. *Chillers WC-1 (3A-SA) and (3B-SB)*

None appear above F-V = 0.001, if at all. Please describe whether their appearance would be expected. If FRANC (software for FPRA analyses) replaces only the probability of an existing basic event with the fire-induced probability, vs. an actual modification of the fault tree logic to include the specific fire-induced basic event, please describe how the importance listed are relevant to the FPRA. Also, please identify the limitations of FRANC in determining importance measures.

Waterford 3 Response

In the FPRA quantification (using FRANC) fire impacts are modeled by setting FPRA basic events associated with fire-failed components and cables to True in the FPRA fault tree and then quantifying the model. The resulting cut sets provide the conditional core damage probability (CCDP) for the fire-affected components and cables failed. Since fire affected basic events are set to True in the fault tree before quantification, these basic events do not appear in the cut sets. Component importances are calculated by merging the cut sets for all sequences for all fire areas into a single cut set file (with fire areas and sequences marked with flags to avoid subsuming similar cut sets from different fire areas and sequences); this merged cut set file is then evaluated for basic event importances. Therefore, the calculated importances are associated with random failures of components and HFEs (human failure events), i.e., non-fire related failures of components not affected by the fire. The importance of fire-affected components is indicated by the CDF contribution of the components at the fire sequence level; the importance measures in PRA-W3-05-007, Appendix F, “Importance Measure Results”, show the importance of the non-fire-affected components. This is a limitation of the FPRA quantification method.

With this FPRA quantification limitation in mind, specific responses for the components listed in the question follow.

a. SSD ESWGR 31AB
This 480 V AC bus appears in the importance results with a F-V of 9.75E-7. This is to be expected because the F-V value is only for the bus’s contribution to non-fire-related failures and because 480 V AC buses have very low random failure probabilities (9.60E-6) compared to other failures (e.g., breaker failures).

b. Panel Aux Panel 4A through 4D
This is not a component, but an electrical cabinet. It is the components associated with cables etc. contained in the cabinet that are mapped to basic events in the FPRA model and thus are included in the importance calculation. The electrical cabinet itself is modeled as an ignition source and container for the associated cables and components (e.g., relays, breakers). Importances are calculated for the components associated with the cables in the cabinet, and these importances are included in the importance results.

c. SWGR 7kV ESWGR 1B
This 6900 V AC bus appears in the importance results with a F-V of 2.17E-6. This is to be expected because the F-V value is only for the bus’s contribution to non-fire-related failures and because 6900 V AC buses have very low random failure probabilities (9.60E-6) compared to other failures (e.g., breaker failures).

d. Control Panel CP-10
This is not a component, but an electrical cabinet. It is the components associated with cables etc. contained in the cabinet that are mapped to basic events in the FPRA model and thus are included in the importance calculation. The electrical cabinet itself is modeled as an ignition source and container for the associated cables and components (e.g., relays, breakers). Importances are calculated for the components associated with the cables in the cabinet, and these importances are included in the importance results.

e. 4kV ESWGR 3B
This 4160 V AC bus appears in the importance results with a F-V of 1.114E-4. This is to be expected because the F-V value is only for the bus's contribution to non-fire-related failures and because 4160 V AC buses have very low random failure probabilities (9.60E-6) compared to other failures (e.g., breaker failures). The F-V for this bus is higher than the F-V values for buses 31AB and 1B because bus 3B directly supports frontline and support systems such as CCW- B, HPSI-B, and EFW-B pumps.

f. Chillers WC-1 (3A-SA) and (3B-SB)
The A and B chillers appear in the importance results with F-V values of 1.59E-2 and 1.35E-2, respectively. This is relatively high, considering that it is only the chillers' importance contribution from non-fire failures that is included in the F-V. In this light, these F-V values are to be expected given the importance of room cooling the multiple mitigating systems.

RAI PRA 40 – 90 Day Response

RAI PRA 41

In the Transition Report, pages 51-55, Table 4-3, Summary of NFPA-805 Compliance Basis and Required Fire Protection Systems and Features, please describe the types of detection that are credited in the areas/zones where required based on the "risk" (R) criterion. For TGB (page 54), please describe the type of suppression system.

Waterford 3 Response

The following list includes the fire areas where detection is credited/required based on risk, and the type of detection.

RAB-1E - Ionization detection
RAB-8A - Ionization detection
RAB-8B - Ionization detection
RAB-31 - Ionization detection
RAB 15 - Thermal detection
RAB 16 - Thermal detection
TGB (FW Pumps and Lube Oil Tank) - Thermal detection

A Deluge system is the type of suppression used in the TGB for areas near the Lube Oil Tank and FW Pumps (ref. FPM-5, FPM-7, FPM-8)

RAI PRA 42

In the Transition Report, pages C-2 through C-672, Attachment C, Table B-3, Fire Area Transition, three sets of inconsistencies are present throughout this table. In one set, a specific component associated with a VFDR is cited as being in the PRA model but not similarly cited in other VFDRs with which it is also associated. For example, in VFDR 17-17, MS-116A is cited as being in the PRA model. However, this same citation is not present in the following VFDRs, also associated with MS-116A: VFDR 3-03, VFDR 7 A-04, VFDR 7C-06, or VFDR 25-02. Please provide clarification.

In the second set, a specific component associated with a VFDR is cited as needing insertion in the PRA model but not similarly cited in other VFDRs with which it is also associated. For example, in VFDR 1-025, MS-119A is cited as needing insertion in the PRA model. However, this same citation is not present in the following VFDRs, also associated with MS-119A: VFDR 7 A-01, VFDR 8A-13, or VFDR 25-04. Please provide clarification.

Also, please identify if any of these were not included in the PRA and if not, explain why not.

In the third set, Table W-2 lists the delta risks associated with a location as epsilon rather than zero, as indicated in Table B-3 (e.g., RAB 70, RAB 23A, RAB 25, RAB 35, RAB 36, Roof E or Roof W), which implies that an evaluation was performed, with the results being negligible (typically below the PRA truncation limit). Please describe if an evaluation was actually performed, or whether these were assumed to be negligible a priori.

Waterford 3 Response

MS-116A is in the PRA model and was evaluated in all FRE calculations where it was identified as potentially being impacted. Fire Area 17 and VFDR 17-17 are different. While MS-116A was mapped to the VFDR, no delta CDF and LERF calculation was performed for Fire Area 17. The fire area is already in compliance with NFPA 805 because no credible fire in the fire area can damage the variant components. As a result, no calculations are necessary, and the delta CDF and delta LERF are zero. In all other listed cases (3, 7A, 7C, and 25) the FRE calculation was performed (with failure of MS-116A in the non-compliant case) and no citation was necessary in the B-3 table.

The VFDR 1-025 included on page C-37 of the B-3 table in the Waterford 3 LAR submittal declares that," the MSIV drain valve (MS119A) is not in the PRA model". While a factual statement, it is not necessary in this portion of the B-3 table. The components PRA status (not modeled) is not relevant to its planned modification (and was thus omitted from similar entries – VFDRs 7A-01, 8A-13, and 25-04). The component was included as a defense in depth variance related to a IN 92-18 failure occurrence only.

Delta risk evaluations were performed for all entries where epsilon is listed as the delta risk value. In these areas the actual delta risk is some very low number between zero and the quantification limits chosen. The impact to the overall results is effectively zero. These were cases where there were PRA modeled components impacted by the VFDRs. However, the quantification values for the compliant and non-compliant cases were identical. Waterford chose to use Epsilon for these cases since the actual result is in truth some non-zero number that is beyond the truncation limit. Since changes were made to the model for the compliant case, the true impact of the change is not actually zero, but is well below any significant value. This is explained in Note 5 of Table W-2 in the LAR submittal.

RAI PRA 43

In the Transition Report, pages S-3 through S-9, Attachment S, Table S-1, Plant Modifications, address the concerns associated with the following modifications:

a. *Page S-3, Items S-1 and S-2. Both modifications are ranked as medium. If the modification is not credited in the FPRA, please describe how it can have an effect other than low (none). Also, if involving circuit re-route, please describe how the possibility of fire-induced effect on the valve via the re-routed circuit is addressed in the FPRA. It would appear that circuit re-route could introduce a fire-induced "failure" in a new location, potentially offsetting the benefit of protecting the valve as per NRC Information Notice (IN) 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire," dated February 28, 1992 (ADAMS Accession No. ML031200481).*

b. *Page S-4, Item S1-3. "These cables were determined to be within the ZOI and could be damaged and fail due to high energy arc failure (HEAF) originating from switchgear SSO-ESWGR-3AB31-S ... This modification is specifically credited from a PRA perspective and reduces the risk of circuit failure due to effects of a HEAF at SSO-ESWGR-3AB31-S that could impact the decay heat removal system cables for valves EFW-228B and EFW-229A" Please describe whether HEAF is the only means by which these cables could be affected from fire. For example, describe whether there is no potential for HGL-induced damage.*

c. *Page S-4, Item S1-4. "... [A] radiant fire-barrier shall be provided to allow the capability of interrupting a fire in the ZOI originating from an HVAC Train-A CHW-MPMP-0001A motor fire or from an HVAC Train-B CHW-MPMP-0001 B motor fire." Please describe whether radiant heat/fire damage is the only means by which fire can cause damage. For example, describe whether there is any potential for HGL-induced damage.*

d. *Page S-4, Item S1-5. Please describe whether the high rank arises solely from the credit in RAB 6, or also from the credit in the other three areas (which appear to be relevant only for non-power operations (NPO), which is not modeled in the FPRA). "This modification is not a result of fire risk evaluations, but assumed in the FPRA model for RAB 6." This is confusing. It appears that the fire barrier was assumed for the FPRA, at least in RAB 6, to reduce fire risk. While there may or may not be need to credit the barrier in the other three areas, it still appears that this modification results from the FPRA.*

e. *Page S-6, Item S1-8. "DG-A and B Fire Areas RAB 15 and RAB 16 heat detectors (120 degrees Fahrenheit (°F) trip set point) will be replaced with new heat detectors with intermediate temperature class (175 to 249 °F) in the two DG rooms. Four TGB fire areas heat detectors (135 °F trip set point) will be replaced with heat detectors with intermediate temperature class (175 to 249 °F) in the four TGB fire areas: turbine lube oil tank, hydrogen seal oil unit, steam generator (SG) feed water pump A, SG feed water pump B systems." Since the new detectors will be less sensitive (i.e., will not respond at lower temperatures), please describe whether this modification is deemed appropriate given the types of fires expected and the need to reduce "false positives". Additionally, please discuss why less sensitive detectors were deemed to be an appropriate modification.*

f. *Pages S-5, S-8, and S-9, Items S1-7, S1-12, and S1-13. Please describe why there is no compensatory measure cited.*

Waterford 3 Response

a. The listed items were ranked 'Medium' conservatively because they impacted the ability of the plant to meet the non-power performance criteria for these zones. The items in S-1 are not in the PRA model. The items in S-2 are in the PRA model. The risk significance of each component is captured in the delta risk numbers associated with each relevant fire compartment. Several of the modeled components also have non-power ops or cold shutdown related functions. Those functions and the desire to correct 92-18 concerns are the reason they have scheduled modifications. None of the modeled component had significant risk impact in the delta risk calculations (many already appear in several fire areas). The re-routing of cables to support the modifications is not expected to be risk significant. Post modification FREs will be completed to ensure NFPA-805 compliance.

b. The specific cables the modification protects are only targets in HEAF scenarios. No other fire scenarios associated with the switchgear (besides HEAF) impact the targets. The physical configuration around and above SSD-ESWGR- 3AB31-S (the area of the modification) precludes the formation of a HGL. Therefore, the HGL evaluation screened out potential HGL impact for a fire in this zone.

c. The HGL analysis considered many factors in screening areas. These factors include: physical makeup of the area, the compartment volume, ignition source type, the presence of flammable gases or liquids, the presence of intervening combustible materials, and the presence of automatic suppression. All fixed ignition sources in RAB 2 (including ones associated with the radiant heat shield modification) were evaluated and none exceeded the screening criteria. So no credible HGL related fire scenarios exist for RAB 2. All scenarios screened without the need to even credit suppression.

d. This modification is the result of the FPRA and the 'high' ranking is based on the credit in only RAB 6. Plant fire protection and PRA personnel implemented the modification based on 'engineering judgment' early in the transition process. The Waterford staff recognized the need based on known separation issues. The modification (3M fire wrap) is credited in the FPRA in RAB6 alone. Fire wrap was not credited in the FPRA in other listed areas.

e. Actuation temperature set-point for heat detectors in RAB 15, RAB 16 and the four TGB areas are required to be intermediate temperature class (175^0 F to 249^0 F) per requirements specified in NFPA-72E-1974 edition, Section 3-4.3. Therefore, this modification is required per NFPA 805 Chapter 3 Section 3.8.2. These heat detectors actuate automatic pre-action and deluge type sprinkler systems. Some of these sprinkler systems have experienced inadvertent actuations due to elevated ceiling temperatures. Thus, indicating the existing temperature set points are to low. Additionally the difference in actuation times for the higher temperature rated detectors is considered insignificant given the combustible liquid hazards in these areas. These type hazards produce high heat release rate type fire that would actuate these suppression systems in a timely manner consistent with NFPA code requirements.

f. Item S1-7: Additional review has determined the existing Chem. Lab Bottle Storage Area located outside the Turbine Building is in compliance with NFPA 50A – 1973 edition code requirements for separation of flammable gas bottles. Therefore, this modification is not needed and Item S1-7 is withdrawn.

   Item S1-12: This is located outside of the power block and does not expose safe shutdown equipment. Additional compensatory measures are not required per the TRM.

   Item S1-13: The quantity of combustibles (plastic caps on 10 – 1 inch diameter pressure relief lines) is insignificant and does not warrant compensatory actions and are not

required per the TRM. Additionally, this is located outside of the power block and does not expose safe shutdown equipment.

RAI PRA 44

In the Transition Report, pages U-3 through U-29, Attachment U, Table U-1, IEPRA Peer Review F&Os, address the concerns associated with the following F&Os:

a. *F&O AS-B3-01. The disposition states that all other phenomena have been addressed. Please describe what "other phenomena" were addressed.*

b. *F&O DA-C10-01; F&Os DA-C12-01 and DA-C12-02; F&O DA-C6-01. "The equipment failure data in question will have an insignificant impact on the FPRA results. In general FPRA failures and quantification results are dominated by fire induced failures which are set to fail due to fire (and not subject to random failure probabilities)." Sequences containing fire-induced failures still may contain random failures whose effect on the fire sequence would be proportionally the same as on the corresponding internal events sequence. Please provide justification why the updates would not be significant for the FPRA and describe whether the conclusion of "insignificant impact" has been at least checked against some potentially affected sequences.*

c. *F&O DA-C8-01. Alternate alignments are not modeled for the full power internal events (FPIE) model according to the F&Os. "The modeling of alternate operating alignments is not expected to significantly impact either the FPRA model, or the IEPRA model it is based on ..." However, some alternate alignments could be important for the FPRA model. Please discuss potentially important alternate alignments for the FPRA model, and their disposition for the LAR. In particular discuss the potential impacts of the alternate alignment of the AS train components, which are assumed in the FPRA to always be aligned to the A train.*

d. *F&O HR-G4-01. "A review of the specific action listed revealed that it has been developed appropriately. This F&O has no impact on the FPRA." Please describe whether the potential effect on timing due to fire phenomena has been considered where the available time limits may have decreased.*

e. *F&O HR-H2-01. (1) "The events discussed meet the definitions of operator recovery actions (RAs) per ASME/ANS RA-Sa-2009, SR HR-H2." One of the requirements to meet the HR-H2 definition is an available procedure and inclusion of operator training, or justification for omitting one or both. If there is no explicit procedure, please describe how this definition is met. (2) "In fact, the worksheet for one action notes the operators do not have enough training or practice to credit the action, although it is given a HEP of 0.1 ... [t]he failure rate in the FPRA is six times higher than is applied in the internal events model." Also, please describe whether the assumed HEP is high enough to reflect the Peer Review concern and justify the lack of training in discussing the appropriateness of the HEP.*

f. *F&O SC-B1-01. The disposition did not address whether or not there was a calculation or reference for the success criteria. During the audit, discussions noted that the success criteria analysis for the drywell and wet well cooling towers has not been updated for the power uprate. Please provide technical justification for the success criteria of these towers for the FPRA. Also, please describe the FPRA success criteria and any impact on it as a result of any updated analyses.*

g. *F&O SC-B1-02. "Hydrogen accumulation in the battery rooms was intentionally neglected in following NUREG/CR-6850 guidance. While the specific battery room scenario in the F&O could increase risk, the amount of risk increase is considered negligible when compared to the hydrogen fires related to hydrogen systems specifically addressed in the guidance." Please describe the effect on the fire risk solely from battery room fire scenarios and whether inclusion of hydrogen fires due to buildup of hydrogen in the battery room proves to be non-negligible for that room.*

h. *F&O SC-B3-02. According to the disposition of SC-83-02, load stripping from the batteries is not credited in the FPRA. If stripping battery loads is a realistic action for fire scenarios, the loads stripped and the time necessary to perform the action may be an important consideration for specific fire scenario sequences. Please discuss the significance of not modeling load stripping for this LAR. In addition, the disposition notes that battery A and 8 have a 1-hour lifetime for non-SB0 scenarios and the basis for this battery time, and how it is included in the FPRA. Please describe whether the timing of any fire-induced scenarios are different from that when the scenarios arise from internal events such that the required time for battery functionality increases.*

i. *F&O SC-B3-01. "Fire induced pipe failure is not considered in the FPRA. Therefore, this F&O does not impact the FPRA." Please describe whether there are any "non-break" loss of coolant accidents (LOCAs) modeled based on their equivalent break size (e.g., spurious opening of valves)? If so, please describe whether any of these could be fire-induced.*

j. *F&O SC-C3-01. "[T]his now appears non-conservative. The lack of unavailability events for the batteries is a model limitation that does impact results. However, due to the very small allowed unavailability time, the events will have a very minor quantitative impact. This finding has no significant impact on the FPRA model or results." Please describe whether the effect been quantified. To the extent a fire-induced scenario involves random failures of batteries and chargers, there could be an impact on the Fire PRA; if so, its significance should be evaluated.*

k. *F&O SY-A12b-01. Please discuss how a 1/3 ratio exclusion for flow diversion is consistent with SR SY-A15. In the disposition, reference is made to "system or train failure." Please describe the context of the system or train that is referred to. Please describe the basis for maintaining that less than a 1/3 ratio does not cause system or train failure and discuss considerations other than the 1/3 ratio in identifying flow diversion pathways and their significance for the FPRA.*

l. *F&O SY-B13-01. "Control room abandonment [in the FPRA] due to loss of heating ventilation and air conditioning (HVAC) is not modeled since it can be easily mitigated." Please describe whether this addresses loss of HVAC with the presence of fire or fire effects.*

m. *F&O SY-C2-02. "The temporary emergency diesel generators (TEDGs) are not credited in the FPRA model, so this finding has no impact on the FPRA mode!." If the FPRA interfaces with the internal events model, please describe whether all credit for use of TEDGs in any internal event sequences triggered by fire-induced initiators is removed.*

<u>Waterford 3 Response</u>

a. In response to this finding, WCAP-16679-P “Accident Sequence Phenomena” was reviewed to determine if any phenomena other than the Steal Line Break (SLB) and Feed Line Break (FLB) impact were not addressed in the current Waterford AS analysis.

The 'other phenomena' referred to in the response refers to all phenomena in WCAP-16679-P that are relevant to the Waterford site. PWROG WCAP-16679-A, "Accident Sequence Phenomena Considerations," developed a list of 18 phenomena that should be considered in Level 1 and Level 2 PRA.

PHENOMENA CONSIDERATIONS

1	Equipment Operation in Beyond Design Basis Conditions
2	Use of Raw Water Systems as Backup Water Supplies
3	Service Water Screen Clogging
4	Containment Sump Debris
5	Loss of Cooling to Critical Equipment
6	Sufficient NPSH for Emergency Core Cooling Recirculation
7	Control Room Habitability
8	Local Operator Actions in Harsh Environments
9	Effect of Loss of Station Air on Air Operated Valves
10	Depressurized Steam Generators – Turbine Driven Emergency Feedwater Pumps
11	Overfilled Steam Generators – Emergency Feedwater Availability
12	Depressurized Steam Generators – Steam Generator Tube Integrity
13	Pressurizer PORVs after Core Uncovery
14	Valves Closing against Large Pressure Differentials
15	Air / Steam Binding
16	Backup Systems for Multi-Units
17	Containment Water Level
18	Mini-Recirculation

The primary focus of the review was to investigate any potential weaknesses the model has with regard to harsh environment phenomena from all high energy pipe failures (secondary impacts associated with the harsh environments).

These data observations are associated with reviews of the number of demands, run hours, and maintenance hours reported in the Maintenance Rule to verify the data is reported properly. Of the referenced findings, only the impact of the finding DA-C6-01 was checked against FPRA results to evaluate potential impact. This F&O deals with the removal of post-maintenance demands and fan speed shifts. It is not clear that the data from the Maintenance Rule includes or excludes these demand failures. If the post-maintenance demands are included in the plant data, then most of the data would revert to NUREG/CR-6928 data. The cutsets were reviewed to determine where plant-specific demand events occurred. The review found that the events occur within the top 1000 cutsets but are not considered dominant risk contributors and would make a small impact of the overall CDF.

b. Regarding F&O DA-C10-01: Waterford used information from plant computer data points to calculate the number of demands and run hours for various systems rather than estimating the number of demands and run hours based on surveillance test procedures. The Entergy/Waterford PRA staff judged this electronically logged data retrieval method to be an acceptable and highly effective method for counting demands and run times. Surveillance test counting is the only method listed in the ASME/ANS Standard, so no other counting method can meet the literal standard. Since the raw data is then Bayesian updated with generic data, the overall impact based on counting methods is further reduced.

   Regarding F&Os DA-C12-01 & -02: The impact would be conservative. Since the maintenance unavailabilities are provided on a monthly basis, it is not clear that

outage hours were removed from the data provided for those months where the generator was not online all month. Therefore, some outage hours might be included in the total number of unavailable hours. Since the denominator is generator online hours, the maintenance unavailabilities could be slightly conservative for some systems or trains.

c. Various alternate alignments were considered during FPRA development. Changes to the alignments associated with the 'swing' pump were examined. These alignments are limited to maintenance cases. The other alignment flag that has some impact on internal event risk and could impact FPRA results is whether switchgear 3AB is powered from switchgear 3A or 3B at the beginning of the event. If the fire impacts the switchgear that 3AB is powered from, the operators would have to swap to the other switchgear. The internal events sensitivity case on the 3AB bus alignment showed that the overall CDF is the same for both alignments. There were some variations to PRA importance measures for equipment based on the variations in alignment, but those are expected when the alignment of the 3AB bus is adjusted.

d. The specific HRA mentioned in this F&O is HHFISOMINP. This HRA is associated with closing the HPSI recirculation lines to the RWSP before sufficient water is diverted from the sump to impact HPSI pump NPSH. The peer team found a discrepancy between the time available to perform the action in the HRA with the time in the success criteria. The revision of the Waterford PRA included a HPSI assumption that stated this time was 1.82 hrs. No supporting calculation could be found to verify this assumption so it was changed to 1 hour. This time discrepancy does not impact this HRA because the new time does not impact the HEP calculated for this action (i.e., cause based analysis is more limiting than the time based analysis). Per the HRA Screening Criteria in PRA-W3-05-003, Figure 5.2.1.-1, the impact on HEP for fire is not impacted by the assumption change.

e. The initial crediting of these actions was based on discussion with operators indicating that the actions would be pursued. These operator insights were the basis for meeting the (a) criteria for HR-H2 (HR-H2 (a) "a procedure is available and operator training has included the action as part of crew's training, or justification for the omission for one or both is provided"). This self-assessment classification of 'met' relied entirely on the 'justification' for the omission of both which was based on the operator insights. The recovery actions in the Waterford PRA were reviewed in detail as part of a NRC CDBI issue documented as CR-WF3-2012-00024. Some of the actions described in the F&O are driven by procedural guidance and could be evaluated using the cause-based decision tree methodology. The assumed HEP value for EHFMANTNR has no impact on the FPRA results since the event does not appear in a single cutset. These HEP related findings (specifically ones regarding HEP credit without procedures) are known limitations of the Waterford Internal Events PRA that the next PRA model update will address.

f. As stated in the RAI and discussed during the audit, the WCT and DCT tower success criteria has not been updated since the Waterford 3 power uprate. This is a known model limitation and an updated cooling tower success criteria will be included in next model update. This limitation is judged to have only a minor impact on the FPRA model and results. While the technical basis for the exact number and combinations of fans is lacking, the system is included in the model. Variations on fan combinations are expected to have a minor impact on results. With the number of fans in the system, system failures are dominated by common cause failures.

g. The battery rooms at Waterford are relatively small and basically only contain the batteries. A fire in the battery room fails all equipment in the room. Hydrogen

accumulation in the battery room is not expected to get high enough to cause a failure of the fire barrier and cause the fire to spread beyond the room (small room with robust concrete block wall). A hydrogen fire due to the buildup of hydrogen has not been postulated. Significant hydrogen buildup would only result from operation of the batteries in response to some other plant transient. An additional battery room fire occurring concurrently with another significant plant transient requiring large loads on the batteries has not been postulated.

h. In the Waterford model, the time available for battery depletion is used only for off-site power recovery and aligning alternate power to the 3AB bus following a failure of the normal power supply. For the fire scenarios, off-site power recovery is not credited for fire-induced LOOP events. The depletions associated with aligning the 3AB bus have a very small impact on the quantified risk. The operator action for aligning 3AB power does not occur with battery depletion in cutsets above 1E-11/yr.

i. The Waterford Fire PRA model includes fire-induced Small LOCA initiators under gate SLOCAFIRE. This gate includes spurious opening of the Reactor Head vent valves, the Pressurizer vent valves, and the Letdown line valves.

j. At Waterford, a battery out of service LCO is 2 hours. With this limiting of an LCO, battery unavailability related work is not scheduled while the plant is operating. A Waterford Corrective Action search found that the Tech Spec associated with battery maintenance/unavailability has only been entered once in the last 10 years. A plant specific battery out of service event would be an order of magnitude or more lower than modeled battery hardware failures. A review of cutsets shows that modeled battery failures are not high in FPRA results. The combined cutset file was reviewed to see the impact battery hardware failures have on FPRA results. These 2E-4 events first show up in cutsets in the 7E-10 range (four orders of magnitude below top cutsets). Plant specific out of service events are expected to be in the same range and have the same (quantitatively insignificant) impact.

k. The context of system and train in the F&O response refers to the system and train definition of an individual plant system. For example: CCW and Service Water usually have multiple trains performing redundant functions (i.e. CCW A train or CCW B train). The 1/3 exclusion rule was investigated to determine how the potential pathway would impact the individual system train (fails CCW train A) or possibly the whole system (path fails all CCW). Most diversion paths only impact the individual train; however, if multiple trains have a common water source, the diversions path could impact multiple train performance. Piping upstream of train splits (common headers) could also impact full system and not just individual trains.

   The 1/3 exclusion rule refers to a flow diversion path size that would cause sufficient flow diversion to impact system or train flow. A flow diversion path size of less than 1/3 of the diameter of the pipe that it is connected to would equate to less than 10% of the flow (with no pressure differential). If the PRA function is a design function of the system, then the function will have at least a 10% design margin. There are two cases where a 1/3 exclusion rule will not apply:

   1. If a sufficient pressure differential exists to allow greater than 10% flow to be lost via a small breach. OR
   2. A PRA-credited system that is credited for something outside its primary design function (E.g., Fire water injection into the primary system).

   Systems were reviewed for both these cases in development of the Waterford 3 model.

The application of the 1/3 rule is consistent with SY-A15. This is primarily true because the system modeling is consistent with SY-A11 (and SY-14). SY-15 is based meeting SY-A11 and SY-A14 – there were no F&Os on 11, 14, or 15 during the Waterford internal events peer review.

l. The potential for a fire outside the MCR (Fire Area RAB 1) to lead to MCR abandonment was evaluated. No such scenarios were found to be credible. This conclusion is based in part on the fact that the dependency of control room HVAC is not explicitly modeled in the PRA model. The Waterford PRA model does not currently have MCR HVAC modeled. The basis for this has been that operators would take local actions (opening doors, installing fans) to ensure loss of room cooling would not threaten the control room habitability. The Waterford/Entergy PRA staff understands that this is a significant model limitation and plan on adding the MCR HVAC dependency to the PRA model in the next model update.

m. The Temporary EDG (TEDG) is not permanently installed equipment at Waterford. The TEDG is brought to the site to allow for an extended diesel outage per Technical Specifications (TS) Action item3.8.1.1.b.2.a. Therefore, no credit can be taken for the TEDG in the baseline PRA or in the Fire PRA. Since the TEDG is not normally on-site, there are no fire-induced initiators that impact the TEDG.

<u>RAI PRA 45</u>

In the Transition Report, pages V-3 through V-16, Att. V, Table V-1, Fire PRA Peer Review F&Os. address the concerns associated with the following F&Os:

a. *F&O ES-A3-02. "This was compensated for by a spurious start of the reactor coolant pumps (RCPs) which would affect the same state in the model." Please clarify this statement.*

b. *F&O ES-B1-01. "The process and documentation did not demonstrate that all high risk-importance components identified in the IEPRA had been considered. Demonstrating and documenting that all high risk importance components identified in the IEPRA had been considered systematically ... [T]he method of identifying additional components from the IEPRA [is discussed] ... The methodology applied and the sensitivity analyses together provide adequate justification that all high risk components have been included in the FPRA." RG 1.200 states that SR ES-B1 requires inclusion of fire risk-significant equipment. Please describe how these were incorporated.*

c. *F&O FQ-A3-01. "The listed section provides details for each case (only 2 scenarios credit self healing). The details in the revised Fire Scenario Report include judgements used in applying the method and calculated time available for given scenarios." Electrical disturbance phenomena due to fires can be quite sporadic, such that reliable estimates of a time line may be difficult to develop. Please describe the results from a sensitivity analysis where credit for self-healing is not taken.*

d. *F&O FQ-B1-01. PRA-W3-05-007, §2.14, documents the results of the convergence evaluation and displays CDF and LERF values at a range of truncation settings. Describe whether convergence is demonstrated and at what level. Please describe how it compares to convergence for the IEPRA.*

e. *F&Os FQ-D1-02, FQ-E1-01, and FQ-E1-02. "A reasonableness review was completed on the internal events LERF results in response to the internal events peer review ... [T]he Summary Report includes a qualitative analysis of uncertainty associated with LERF results (i.e., a reasonableness review)." F&O UNC-A 1-01. "Additionally, section*

3.2 of the Summary Report includes a qualitative analysis of uncertainty associated with LERF results (i.e., a reasonableness review)." FQ-D1-02. Please describe what was done for the reasonableness review and provide the results. As stated by RG 1.200, the SR QU-E3 for HLR-QU-E for Internal Events, to which these findings appear related, requires an "estimate" of the uncertainty intervals for both CDF and LERF. Please describe to what extent this "reasonableness review" provides such estimates. The disposition seems to suggest that only a qualitative evaluation was performed.

f. *F&O FSS-D7-01. "Section 8.1 of PRA-W3-0S-006, documents a plant specific analysis of fire suppression system failure probabilities. This analysis includes a review of maintenance history and unavailability. This section provides adequate details to the meet the Category II SR requirements for FSS-D7." Please describe whether any outlier behavior was observed. If so, use of generic estimates does not satisfy CC-II.*

g. *F&Os HR-A2-01 and HRA-A4-01. "However, no HRA calculation sheets could be located in the FPIE or in the FPRA documentation ... No documentation could be found ..." It appears this could be more than just a documentation deficiency, in that the required analyses may not have been available for the Peer Review. Please clarify the disposition of these F&Os.*

h. *F&O HRA-C1-01. "The detailed analysis of the fire affected HFEs should be developed more addressing the fire effects on the action. Particular attention should be focused on the required instrumentation." Please describe whether the fire effects were addressed, but just not documented, or were they not addressed at the time of the Peer Review and how it was determined that no new cable instruments were identified that needed to be added. Please describe the current status of the disposition of this F&O.*

<u>Waterford 3 Response</u>

a. This statement in the F&O was intended to acknowledge that the failure to trip the RCP was modeled as a spurious start of the RCP. Both failure modes, due to loss of DC (given that the RCP is initially operating), have the same impact.

b. Several steps were completed to ensure all risk significant components (IE risk significant and/or fire risk significant) were included in the model. This was primarily accomplished in following the NUREG/CR-6850 methodology, specifically Task 2 Fire PRA Component Selection. Specific tasks to achieve this goal included:

   - Mapping IEPRA events to FPRA events
   - Evaluating SSEL equipment list to add or exclude (with justification) to FPRA
   - Adding events or equipment whose spurious or fire induced operation could impact success of a PRA credited function. (including MSO expert panel review of the effort)
   - Adding events or equipment whose spurious operation could induce inappropriate or unwanted actions by plant operators.

   The component and cable selection effort represents a systematic effort to include all reasonable risk significant components in the FPRA model.

c. The WSES FPRA included the treatment of spurious actuation of plant equipment resulting from hot short (intra and inter) cable failures. In only very limited instances was credit for self-healing of AC power circuits applied using the treatment described in FAQ 08-0051. The specific cases where credit was applied are summarized below:

Basic Event ID	Function	Fire Scenario
RSVRC1014R	Rx System Vent Valve	RAB5-T001, T002
RSVRC1015R	"	RAB6-T002, T003, T004
RSVRC1017R	"	RAB5-T001, T002
RSVRC3183R	Pressurizer Heat Vent Valve	
RSVRC3184R	"	RAB6-T002, T003, T004
RSVRC3186R	"	
UMCSV106AN	AH-25A Damper	RAB7A-E002, RAB7B-E001, T001
UMCSV106BN	AH-25B Damper	

The application of the self-healing factor is consistent with the guidance provided in Supplement 1 to NUREG/CR-6850 and recognizes that no credit should be applied for DC power circuits or cases where shorting to ground and consequential loss of power would not result in the component failing into the desired safe state.

With respect to the specific NRC information request, a sensitivity treatment was performed where no credit for self-healing was applied. The results are summarized below:

Fire Scenario	Current CDF	Sensitivity Cases	
		0.10	1.0
RAB5-T001	3.3E-7	3.3E-7	4.4E-7
RAB5-T002	4.8E-8	6.7E-8	1.5E-6
RAB6-T002	3.6E-7	3.6E-7	5.1E-7
RAB6-T003	9.9E-8	1.0E-7	1.9E-7
RAB6-T004	6.3E-8	6.4E-8	1.2E-7
RAB7A-E002	1.1E-7	1.1E-7	1.1E-7
RAB7B-E001	4.9E-7	4.9E-7	4.9E-7
RAB7B-T001	1.3E-7	1.3E-7	1.3E-7

d. The truncation values used in the FPRA quantifications are 1E-09 and 1E-10 for CDF and LERF, respectively. The table below shows the results of the test for convergence. As shown in the table below, the selected truncation is one order of magnitude lower than needed. It should be noted, that the convergence test was performed on a version of the Fire PRA that has changed somewhat as the result of final comment resolution and Peer Review issue resolution. Although some changes have occurred, there have been no substantive changes that would warrant the reverification of these results. (From PRA-W3-05-007)

Truncation	CDF	% Chg	LERF	%Chg
1.00E-07	3.27E-05		NA	NA
1.00E-08	3.37E-05	3%	NA	NA
1.00E-09	3.40E-5	1.0%	7.18E-07	-

Truncation	CDF	% Chg	LERF	%Chg
1.00E-10	3.42E-5 .	0.4%	7.34E-07	2.2%
1.00E-11	3.42E-5 .	0.1%	7.40E-07	0.8%

Based on the results shown above, convergence begins for CDF and LERF with a truncation of 1E-7 and 1E-9, respectively. The truncation that has been selected for CDF and LERF are 1E-9 and 1E-10, respectively.

e. Only a qualitative LERF uncertainty analysis was completed. A quantitative (parametric uncertainty analysis of CDF results) uncertainty analysis was completed for CDF. Appendix D of the Waterford FPRA Summary Report (PRA-W3-05-007) provides the details on sources of uncertainty for each task in the FPRA development. Only a qualitative uncertainty is provided for LERF results which are in 3.2 of the Summary Report.

f. The conclusion of the plant specific reliability analysis on fire suppression equipment was that there was no outlier performance/history. This conclusion supports the use of generic data (and meeting the Category II SR requirements for FSS-D7).

g. The first issue is associated with only event RHFPUMPOFP, which did not have a detailed HRA evaluation for the FPIE because it did not appear in any cutsets. The screening value for this event was increased for FPRA. Since this event did not occur in the FPRA cutsets until about 3.8E-11/yr, the screening value was considered adequate.

   No operator actions from the Off Normal Procedures (OP-901-502 and OP-901-524) were added in the FPRA quantification.

   In response to this F&O, Waterford added documentation of simulator observations and control room practices during fire scenarios in Appendix E of PRA-W3-05-003. However, these documentation enhancements did not impact overall HRA results used in the FPRA.

h. Appendix D of the Fire PRA Quantification Model Preparation and Database Development report (PRA-W3-05-003, Revision 0) discusses the assessment considerations based on NUREG-1921 Appendix C that were made for the detailed HEP calculations. These considerations include fire impacts. Additional operator interviews were conducted post-peer review and any additional information was reflected in the detailed calculations recorded in Appendix D of the post-peer review version of this report.

   Table A-1 within Appendix A of the post-peer review version of this report contains expanded detail on fire-protected cues gathered in Waterford operator interviews conducted subsequent to the peer review. Cues protected from fire impacts include:

   - control room indications that are on the SSEL,
   - environmental cues which do not require instrumentation such as changes in CR lighting caused by a loss of offsite power, and
   - procedure steps located within procedures that are initiated based on indications that are protected from fire impacts.

   Cable information was not required for operator actions having the above-listed fire-protected cues. Note that cables for the SSEL indications were included in FRANC databases for use in quantifying the model.

RAI PRA 46

In the Transition Report, page W-3, Att. W, §W.2, Increase in CDF and LERF, the cited total increase in LERF is 6.6E-7/Yr. Two proposed modifications are cited to decrease LERF by 3.5E-7/yr. Please explain why the net total increase in LERF is cited as 1.1 E-7/yr instead of 3.1 E-7/yr (6.6E-7 -3.5E-7 = 3.1 E-7).

Waterford 3 Response

The cited total increase in LERF (with modifications credited) of 1.1E-7/yr is correct. The total increase in LERF of 6.6E-7/yr on page W-3 of LAR is incorrect. The correct value is 4.6E-7/yr. This is the number listed in Table W-2 (page W-16).

LERF without modifications (4.6E-7) – LERF with modifications (3.5E-7) = 1.1E-7/yr

RAI PRA 47

In the Transition Report, on pages W-13 through W-16, Attachment W, Table W-2, Fire Waterford 3 Fire Area Risk Summary, RAB 7C is the only case where delta-LERF exceeds delta-CDF. Explain why. The totals for additional risk of RAs (cited as 3.6E-6 for CDF and 8.8E-8 for LERF) sum only to 3.2E-6 and 3.8E-8 (with contributions only from RAB 1 [AIE], RAB Sand RAB 6). Please explain this discrepancy.

Waterford 3 Response

The sums listed in Attachment W are errors/typos. The correct values are is 3.2E-6 for CDF and 3.8E-8 for LERF(as noted in question) for the net increase in risk (LERF) following planned modifications.

For the case noted (RAB 7C) net change in LERF is greater than CDF due to the non-compliant case cutset file of the LERF scenario E002 having one additional cutset in it that involves an ISLOCA sequence with failure of SI-401B. The compliant case does not contain this cutset and this was the top cutset of the non-compliant version of E002 for LERF. The sequence is based on the removal of SI-405B from the fire area. Valves SI-405B and SI-401B are two valves in series with the potential (if both fail open) to lead to an ISLOCA (and LERF). In the compliant case this is not possible with the SI=405B valve set to True. These two valves failing open are the top cutset in the non-compliant case.

RAI PRA 48

In W3F1-2012-0005, page 5 of 7, SO 3, Aggregate Impact on Fire Risk Results, it appears from the discussions during the audit that some credit in now being taken for proposed modifications that was not being taken in the LAR as a result of being prompted for the methods Sensitivity Analysis. Describe whether any of the reported risk or delta-risk results changed as a result (Le., without the methods Sensitivity Analysis, but with the "new" credit). Please provide updated LAR information if the LAR is not longer accurate.

Waterford 3 Response

Of the modifications listed in attachment S, only one S1-5 (a 3M fire wrap in RAB 2, RAB 5, RAB 6, and RAB 17) is directly credited in the Fire PRA model. This is true for both the results

documented in the LAR and the methods sensitivity analysis. The delta risk values listed in Attachment W of the LAR are unchanged as a result of the W3F1-2012-0005 analysis. It was intended to demonstrate how variation in methods may impact the results. It was not intended to replace the results of Attachment W.

The modification credited is a fire wrap that is credited in RAB 6 only. The wrapped cables in RAB 6 are assumed protected from fire in the base FPRA model. The wraps in RAB 2 and RAB 5 are not credited in the PRA model (though they are considered part of the same modification). This modification is already physically installed. The modification is not technically complete only due to the modification closeout documentation not being complete. None of the other modifications listed in Attachment S are credited in the FPRA model.

RAI PRA 49

For SRs FSS-E3 and -H5, without an uncertainty or sensitivity analysis, even a qualitative estimate must assure that the risk and delta-risk results are representative, and conservative as needed where uncertainty may be large, even if not readily quantifiable. Please provide the referenced discussions in PRA-W3-05-006 and -007.

Waterford 3 Response

SR FSS-E3:

The treatment of uncertainty and sensitivity is primarily limited to those fire scenarios where the refinements described in Tasks 10 (Circuit Failure Mode and Likelihood Analysis), 11 (Detailed Fire Modeling), and 12 (Post Fire HRA) were applied. This is because these scenarios are those that otherwise would have been notable risk contributors. Other fire scenarios that were not subject to aggressive refinements are expected to maintain a degree of conservatism so that their treatment would more closely resemble that of an ‘upper bound’ analysis.

Ignition source counting is an area with inherent uncertainty; however, the results are not particularly sensitive to small changes in ignition frequency. First, the impact of the uncertainty associated with ignition source counting tends to be random in nature. Second, it generally takes a significant change in the number of ignition sources to have an order of magnitude difference on an individual result. Given the methodology for allocating frequency, two similar plants with a significantly different overall population of Bin 15 electrical cabinets would also report different individual cabinet fire frequencies. However, differences in cabinet fire frequency by itself are not enough to characterize one result as conservative and the other non-conservative without considering potential differences in cabinet size, cable loading, and functional importance. If only those ignition sources that could lead to risk significant scenarios were counted, the scenario frequencies would increase but would not be consistent with industry experience. Conversely, “over counting” non-risk significant ignition sources would lead to lower individual scenario frequencies but is perhaps more representative of industry experience since most plant fires are not risk significant. While the uncertainty associated with ignition source counting is generally random, ignoring plant structures (and their ignition sources) determined to have no risk impact during the plant partitioning task tends to bias the ignition source frequencies in the conservative direction. Consequently, the uncertainties associated with plant partitioning and ignition source counting tend to balance out and are minor changes which will not significantly alter the results (exclusion of areas with no plant impact in the plant partitioning task reduces the number of components to which the plant’s total ignition frequency is allocated).

Bayesian updating of the NUREG/CR-6850 bin frequencies has been performed to reflect the change in likelihood of fire given the occurrence or non-occurrence of related fire events for the plant.

Another biased uncertainty in the conservative direction pertains to the cable selection process and the assignment of circuit failure probabilities. The underlying circuit analysis was that used in the safe shutdown analysis. Since the safe shutdown analysis focused on demonstrating the availability of one safe shutdown path, circuit analysis for other paths which contribute to overall plant risk was typically not performed. In turn, the fire PRA typically linked the entire set of cables to every basic event associated with a given component. A circuit failure probability was typically applied only to specific basic events associated with spurious operations which resulted in a higher risk contribution. Therefore, in instances where only those cables that could not lead to spurious operation were damaged, an additional conservatism is applied in that the spurious operation related basic event was assigned a circuit failure probability even when the associated failure mode (e.g., hot short) was not credible for that scenario.

Table D-1, reproduced below from PRA-W3-05-007, provides qualitative evaluation of the uncertainties and sensitivities for the Fire PRA tasks (from NUREG/CR-6850).

TABLE D-1
UNCERTAINTY AND SENSITIVITY MATRIX

TASK NO.	SOURCES OF UNCERTAINTY	SENSITIVITY OF THE RESULTS TO THE SOURCE(S) OF UNCERTAINTY
1	This task poses a limited source of uncertainty beyond the credit taken for boundaries and partitions.	During scenario development, the zone of influence was not limited to the physical analysis unit boundary. If the zone of influence included targets in adjacent fire zones, these targets were also included, regardless of their fire zone location. In addition, the multi-compartment analysis further reduces this uncertainty by addressing the potential impact of failure of partition elements on quantification.
2	This task poses perhaps the highest potential for error if not uncertainty. The mapping of basic events to components requires not only the consideration of failure modes (active versus passive) but an understanding of the Appendix R functions not previously considered risk significant in the FPIE model. When performed correctly, the only uncertainty is related to the MSO process.	The potential for uncertainty is reduced as a result of multiple overlapping tasks including the MSO expert panel. Additional internal reviews and the change evaluation process further reduce uncertainty in this task.

TASK NO.	SOURCES OF UNCERTAINTY	SENSITIVITY OF THE RESULTS TO THE SOURCE(S) OF UNCERTAINTY
3	No treatment of uncertainty is typically required for this task beyond the understanding of the cable selection approach (i.e., mapping an active basic event to a passive component for which power cables were not selected). Additionally, PRA credited components for which cable routing information was not provided represent a source of uncertainty (conservatism) in that Y3 components could be assumed failed unnecessarily.	The limited number of Y3 components (most active components credited in the fire PRA were included in the PDMS database) as well as the crediting by exclusion of Y3 components (where justified) helps to reduce unnecessary conservatism. Sensitivity quantifications were performed in which the Y3 components were assumed to be available (as opposed to damaged) for all fire scenarios. The results of these sensitivity runs indicated that the total CDF was not significant. The actual configuration in which the Y3 components are lost in some fire zones but not all fire zones would result in a smaller reduction in CDF. Therefore, the impact of this uncertainty is not considered particularly significant in light of credit for these components by exclusion where their loss was creating a significant impact on the risk of a given fire scenario.
4	Qualitative screening was not performed; however, structures were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the fire PRA were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip.	In the event that a structure which could lead to a plant trip was excluded incorrectly, its contribution to CDF would be small (with a CCDP commensurate with base risk) and would likely be offset by inclusion of the additional ignition sources on the reduction of other scenario frequencies. A similar argument can be made for ignition sources for which scenario development was deemed unnecessary.
5	A reactor trip is assumed as the initiating event for all quantification. This is somewhat conservative since not all fires postulated will result in a plant trip.	FPIE and fire PRA peer reviews (including the F&O resolution process), internal assessments, and the change evaluation process are useful in exercising the model and identifying weaknesses. Generic MSO scenarios identified by the industry subsequent to the initial WSES MSO Expert Panel were added to the fire PRA using the same methodology employed in addressing MSO scenarios identified in the original MSO Expert Panel.

TASK NO.	SOURCES OF UNCERTAINTY	SENSITIVITY OF THE RESULTS TO THE SOURCE(S) OF UNCERTAINTY
6	Ignition source counting is an area with inherent uncertainty; however, the results are not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the frequency values from NUREG/CR-6850 which result in uncertainty due to variability among plants along with some conservatism in defining the frequencies, and their associated heat release rates, based on limited fire events and fire test data.	A Bayesian update process for events after 2000 was applied to the generic frequencies taken from NUREG/CR-6850. With some conservatism in the ignition frequency data, combined with conservatism in their associated heat release rates, the uncertainty in Task 6 is biased in the conservative direction.
7	Other than screening out potentially risk significant scenarios (ignition sources), there is no uncertainty from this task on the fire PRA results.	Quantitative screening is limited to refraining from further scenario refinement of those scenarios with a resulting CDF/LERF below the screening threshold. All of the results were retained in the cumulative CDF/LERF.
8	The approach taken for this task included: 1) the use of generic fire modeling treatments in lieu of conservative scoping analysis techniques and 2) limited detailed fire modeling was performed to refine the scenarios developed using the generic fire modeling solutions. The primary conservatism introduced by this task is associated with the heat release rates specified in NUREG/CR-6850.	The employment of generic fire modeling solutions did not introduce any significant conservatism. Detailed fire modeling was performed on those scenarios which otherwise would have been notable risk contributors. Detailed fire modeling was only applied where the reduction in conservatism was likely to have a measurable impact. The NUREG/CR-6850 heat release rates introduce significant conservatism given the limited fire test data available to define the heat release rates and the associated fire development timeline. Quantification of the impact of this conservatism is pending further evaluation of the NUREG/CR-6850 methodology by the MOU group.
9	Uncertainty considerations are limited to errors in circuit failure analysis where a cable was deemed incapable of causing loss of a particular function credited in the fire PRA. Similar to Task 2 (with the exception of the MSO process), this task has no associated uncertainty when performed correctly.	Circuit analysis was performed as part of the Appendix R Analysis. Refinements in the application of the circuit analysis results to the fire PRA were performed on a case by case basis where the scenario risk quantification was large enough to warrant further analysis. Therefore, the uncertainty/conservatism which remains in the evaluation is associated with scenarios which do not contribute significantly to the overall fire risk.

TASK NO.	SOURCES OF UNCERTAINTY	SENSITIVITY OF THE RESULTS TO THE SOURCE(S) OF UNCERTAINTY
10	The uncertainty associated with the applied conditional failure probabilities poses competing considerations. On the one hand, a failure probability for spurious operation could be applied based solely on cable scope without consideration of less direct fire affects (e.g., a 0.3 failure likelihood applied to the spurious operation of an MOV without consideration of the fire-induced generation of spurious signal to close or open the MOV). On the other hand, a failure probability for spurious operation could be applied despite the absence of cables capable of causing spurious operation in that location.	Circuit failure mode likelihood analysis was generally limited to those components where spurious operation could not be caused by the generation of a spurious signal. This approach limited the introduction of non-conservative uncertainties. For the 'simple' cases, the potential exists for assuming a failure likelihood greater than 0 versus 0 (or random) failure likelihood in some areas where the cables capable of causing spurious operation are not located. Additional refinement to this approach was performed, as necessary, on risk significant scenarios. So the application of circuit failure probabilities is considered to have minimal impact on the results.

TASK NO.	SOURCES OF UNCERTAINTY	SENSITIVITY OF THE RESULTS TO THE SOURCE(S) OF UNCERTAINTY
11	The primary uncertainty in this task is in the area of target failure probabilities. Conservative heat release rates may result in additional target damage. Non-conservative heat release rates would have an opposite effect. Credit for fire brigade response, and detection are limited to the treatment used for vented panels and the hot gas layer evaluation. For fire zones with detection, the vented panel treatment credits a 30 minute timeframe prior to the fire impacting targets outside of the panel. Only 15 minutes is considered to be available for fire brigade response if the detection system fails. These numbers are to be realistic and somewhat conservative. For the hot gas layer evaluation a response time of 20 minutes is assumed for the fire brigade to intervene and limit further development of a hot gas layer. This timeframe is consistent with results of actual brigade drills and is somewhat conservative in that many fire scenarios would result in a shorter response time.	Detailed fire modeling was performed only on those scenarios which otherwise would have been notable risk contributors and only where removal of conservatism in the generic fire modeling solution was likely to provide benefit either via a smaller zone of influence or to credit automatic suppression. Fire modeling was used to evaluate the time to abandonment for control room fire scenarios for a range of fire heat release rates. The analysis methodology conservatism is primarily associated with conservatism in the heat release rates specified in NUREG/CR-6850. Quantification of the impact of this conservatism is pending further evaluation of the NUREG/CR-6850 methodology by the MOU group. Additional refinement of the fire scenarios can be pursued using multi-point analysis of the heat release rates as opposed to the use of a bounding fire for most scenarios. A review of the generic fire modeling treatment summary zone of influence data, however, indicates that the reduction in zone of influence for smaller fires is not significant. The potential for this slightly reduced zone of influence to reduce the consequences associated with the smaller fire is very small. Without a reduction in consequences a multi-point treatment of the heat release rate curves would have no impact on results. Additional fire modeling can be pursued in areas of high risk. A review of the targets of concern for the high risk scenarios with respect to the associated ignition sources indicates that the likelihood of detailed fire modeling demonstrating survival of key targets is very small. Further refinement in the above areas is considered premature given that the heat release rate assumptions and fire development timeline is currently under industry and regulator review. Further evaluation of the potential benefit of multi-point heat release rate curve scenario definition and detailed fire modeling should be evaluated once further refinements in the heat release rate and fire development timeline profile are agreed upon.

TASK NO.	SOURCES OF UNCERTAINTY	SENSITIVITY OF THE RESULTS TO THE SOURCE(S) OF UNCERTAINTY
12	Human error probabilities do not represent a potentially large uncertainty for the fire PRA given the relative few HFE of high importance in the base model. Since many of the HEP values were adjusted for fire, the joint dependency multipliers developed for the FPIE model also represent a potential for introducing a degree of conservatism.	Conservative HEP adjustments were made to the nominal HEP values used in the FPIE model then revisited to address unique fire considerations. Screening HEP values were used for recovery actions which were not currently modeled in the FPIE PRA.
13	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	Seismic fire interaction has no impact on fire risk quantification.
14	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit.	Since the fire PRA solves for CCDP (prior to the application of frequency) at a truncation limit of 1E-9, there should not be a significant truncation contribution. This truncation limit is several orders of magnitude below the typical CDF value calculated.
15	This task does not introduce any new uncertainties but is intended to address how uncertainties may impact the fire risk.	N/A
16	This task does not introduce any new uncertainties to the fire risk.	The documentation task compiles the results of the other tasks. See specific technical tasks for a discussion of their associated uncertainty and sensitivity.

SR FSS-H5:

The Hughes Associates Generic Fire Modeling Treatments used in the Waterford 3 Fire PRA offers a means for incorporation of fire modeling into the fire PRA in a manner that eliminates the need for separate scenario specific analyses which require significant effort for configuration control, review and update. The Generic Fire Modeling approach incorporates a level of conservatism which can impact overall results. However, given that the input parameters (ignition frequency and heat release rate) to the fire model, Generic or Scenario Specific, contain significant conservatism, efforts to reduce conservatism by applying detailed fire modeling prior to refinement of the input parameters would result in significant effort in modeling and configuration control which will potentially require significant revisions once the input parameter refinements are available.

The following refinements are not directly addressed in the Generic Fire Modeling Treatments, the basis for acceptability of eliminating these refinements from the fire PRA fire modeling is discussed below:

> Multi-Point Heat Release Rate Treatment The factors specified for vented panels provide a multi-point treatment for vented panels based on a split fraction developed from the EPRI Fire Events Database that specifies the fraction of fires impacting only the ignition source panel versus those fires which impact targets within the zone of influence of the panels. The use of other similar factors, for high voltage panels, provides a means for further expanding this type of multi-point treatment. This approach not only provides

a more definitive means of differentiating between significant and limited fires but provides a means that will not be impacted by potential future refinements in ignition frequency and heat release rate. The severe fire scenario impacting external targets typically uses the current 98th percentile heat release rates.

Incorporation of Fire Growth Curve The use of a split fraction for fires impacting the ignition source panel versus those fires impacting targets within the zone of influence provides a fire events based means for addressing fire growth with respect to impact on internal versus external panel targets. Since the fire events data incorporates actual fire growth and suppression data as opposed to correlation of generic fire ignition frequencies to unrelated heat release rates, the methodology used is significantly more realistic and likely less conservative than the use of heat release rates.

Detailed Fire Modeling The use of detailed fire modeling may be useful once realistic ignition frequencies and heat release rates are derived. The use of detailed fire modeling with conservative input parameters applies refinements to gross conservative approximations and will not yield realistic results. The benefits associated with detailed fire modeling must also be weighed against the complexities of configuration control requirements for maintaining a complex fire modeling analysis for many scenarios throughout the plant. The use of fire modeling once input parameters are refined and for specific fire scenarios where the conditions support a less conservative outcome than the use of the Generic Fire Treatments would be appropriate. However, at the current state of fire PRA input parameters the use of across the board fire modeling would not provide a significant improvement in fire PRA results.

RAI PRA 50

Fire Areas 7 and 8 are partitioned into fire compartments (A, B, C, ...) some of which have partial height wall boundaries. Partial height boundaries are explicitly excluded as a partition boundary in NUREG/CR-6850. Please clarify how these are treated in the FPRA in light of the NUREG/CR-6850 guidance.

Waterford 3 Response

The existence of the partial height walls was documented in the Report PRA-W3-05-001 Waterford FPRA Partitioning & Fire Frequency Development Table 2-1 Notes 4 and 5. For consistency with the cable routing information the PAUs were designated similar to the original App R fire zones, that is, these fire zones are used as an analysis convenience. When evaluating these subdivided fire areas (7 and 8), the potential impact of fires within one zone on the adjacent is addressed in scenario development by inclusion of all targets in the ZOI regardless of fire zone location (i.e., target on the other side of the wall but still within the zone of influence would be included.) In addition, the multi-compartment assessment addresses the potential effects of hot gas migration. Further, given the potential for interactions with the other PUAs, these zones could not have been, and were not, screened solely on the bases in impacts associated only with the PAU.

RAI PRA 51

The FPRA peer review noted in L TR-RAM-11-003 that some F&Os were designated as not applicable (N/A). With respect to FSS-G5 and FSS-G6, please describe the important contributors to screening out all MCA scenarios. Also, describe how the summation of the screened MCA scenarios compare to the Regulatory Guide 1.200 resolution for ONS-C1.

Waterford 3 Response

The following is quoted from the Waterford 3 Fire PRA Peer Review Supporting Requirement Assessment Summary:

"FSS-G5 - Per Report 0247070001.05 (PRA-W3-05-005), Rev. 0, all MCA scenarios were screened. No active fire barriers were identified.

FSS-G6 - Per Report 0247070001.05 (PRA-W3-05-005), Rev. 0, all MCA scenarios were screened. No other multi-compartment scenarios were analyzed."

The conclusion of the Waterford 3 MCA analysis was that there were no multi-compartment issues (Section 3.4 of MCA analysis). With all compartments (and active barriers) screening, FSS-G5 and FSS-G6 were classified as N/A.

The NRC position clarification on QNS-C1 provided in Regulatory Guide 1.200 relates to screening criteria based on quantification results. The clarification addresses the point that screening should be based on Fire CDF and LERF (as opposed to internal events). No MCA quantification results exist at Waterford 3, so this clarification has no impact on the Waterford 3 peer review assessment.

RAI PRA 52

Please clarify if IN 92-18 issue resolution is still under review.

Waterford 3 Response

IN 92-18 issues are no longer under review. Valves with IN-92-18 concerns have planned modifications to eliminate the IN 92-18 issues and are included in the S1-1 and S1-2 lists.

RAI PRA 53

For Components:

a. *VFDRs show both low pressure safety injection (LPSI) and high pressure safety injection (HPSI) are dependent upon 125 volt direct current (VDC) control power from DC bus/panel DC-EPDP-B-DC (e.g., VFDR 8C-16 and VFDR 8C-17).Please describe whether there is also a potential affect on its associated alternating current (AC) bus/panel ID EUPSMB. If there is a potential fire-related affect, describe the basis for not including it in the FPRA.*

b. *With respect to VFDRs 8C-10 and 8C-11, describe whether fire damage to the EFW-228B valve power and control cables affects both valves EFW-228B and EFW-229A. If so, discuss why it occurs and the failure mode(s) of the valves, or provide clarification.*

Please describe whether the RAs for these valves in Table G of the LAR apply in this case.

c. *Please clarify for the FPRA if solenoid operated valve (SOV) dependencies are logically linked to the SOV basic event such that failure of a dependency is propagated to the valve. If not, describe how failure of a dependency for a SOV is handled in the FPRA.*

Waterford 3 Response

a. The impact of VFDRs 8C-16 and 8C-17 were addressed in the RAB 8C FRE in terms of changes in CDF and LERF. The FPRA model contains mapped events representing the loss of the LPSI and HPSI pumps as a result of a fire impacting the dc panel. There is no potential fire related affect of EUPSMB in RAB 8C. This 120 VAC bus is included in both premature RAS (gate AHPRERASB) and RAS failure logic (gate A001). This component is also included in the FRANC database (ID-EUPS-MB).

b. Both valves EFW-228B (8C-10) and EFW-229A (8C-11) have power and control cables routed in the same area (above and within the ZOI of a 480V AC switchgear). There are six PRA basic events associated with these VFDRs. For each; valve events for 'transfer open', 'fail to open' and 'transfers closed' are mapped in the RAB 8C FRE.

   None of the RAs in Attachment G specifically apply to these variances. None of the RAs listed in Attachment G are for fires in RAB 8C.

c. Solenoid valve dependencies are logically linked (in the fault trees) in the Waterford 3 model. Generally SOVs have 480 VAC, 120 VAC or 125 VDC dependencies. However, if the SOV fails closed following a loss of power and the only PRA function is for the SOV to close, then the power dependency would not be modeled for the valve. Note: Several solenoid valves are used to support air-operated valves. The PRA does not need to model these SOVs because they are included in the AOV component boundary (as discussed in NUREG/CR-6928).

RAI PRA 54

System Modeling

a. *Table B-3 of the LAR for Fire Area RAB1 identifies the following for Vital Auxiliaries (mechanical):*

   > *... component cooling water (CCW) Train B with Train B dry cooling tower (DCT) auxiliary component cooling water (ACCW) Train B with Train B DCT*

   During the audit, discussions noted that "DCT" was a typographical error and it should be wet cooling tower (WCT) for the ACCW system. Please correct this error.

b. *Please describe whether the emergency feedwater system (EFW) is taking suction from the WCT and if there is any impact on the success criteria for CCW or ACCW. If so, describe how it is modeled in the FPRA.*

c. *Please describe how sufficient EFW inventory is maintained for extended hot shutdown and if EFW inventory for 24 hours is modeled in the FPRA. Please describe if potential fire impact to aligning EFW to the WCT basin is considered in the FPRA. For example, describe if the fire's location can prevent manually opening the ACCW valves for EFW suction to the WCT.*

d. *During the audit, discussions noted that room cooling was modeled through consideration of fire impacts on room cooler cables. Please describe whether the fire impact on room cooler cables was modeled for risk-important rooms in the FPRA. In addition to potential impact on cables, consider quantitatively or qualitatively the significance of fire impact on room cooling as it affects cooling to pumps due to room heat-up. Please provide a summary of the assessment for risk-important pumps in the FPRA.*

Waterford 3 Response

a. The typo of DCT which should be WCT for the ACCW system will be corrected.

b. Emergency Feedwater - ACCW provides a backup to the CSP as a water source for EFW pump suction. Each train of ACCW is connected to EFW through two normally locked closed manual valves, ACC-114A(B) and ACC-116A(B). Additional ACCW inventory is available through the WCT Basin. The WCT Basin water may also be aligned if the inventory of the Condensate Storage Pool (CSP) is depleted during an accident. However, only one train of ACCW may be aligned to EFW. The other train must remain intact for use as part of the ultimate heat sink. An ACCW pump must be running in the train supplying EFW. This is in the PRA model logic. ACCW is modeled as a support system for EFW through the WCT.

c. The normal suction source for EFW is the Condensate Storage Pool (CSP). This source has sufficient inventory for approximately 10 hours. Following CSP depletion, the operators can align makeup to the CSP from the Condensate Storage Tank (CST) or swap EFW suction to the WCT basin. Either of these sources would provide sufficient inventory to extend hot shutdown (well beyond 24 hours). The PSA model includes these sources for EFW suction. The Waterford Fire PRA includes scenarios that impact the alignment of WCT to provide EFW suction.

d. Fire impacts on room cooler cables are modeled for risk important rooms in the Waterford 3 FPRA model. Many risk important systems and components in the Waterford PRA (and FPRA) model have room cooling dependencies modeled. The rooms with HVAC dependencies are listed below. These dependencies are explicitly modeled in the logic such that a failure of the room cooling (or fire damage to room cooling cables) leads to failure of the pump and/or system function due to room heat-up.

Loss of Room Cooling in the Waterford 3 PRA affects the following:

- Safeguards Rooms (HPSI, LPSI, Containment Spray Pumps)
- CCW Pumps
- Diesel Generators
- EFW Pumps
- Switchgear Rooms

RAI PRA 55

Sequences

a. *Please discuss the flow diversion paths which were considered for LERF modeling, and, the basis for their dispositions.*

b. *A screening criteria for LERF in PRA-W3-05-002, "FPRA Component and Cable Selection Report," is that a pathway is water solid. Please discuss how has the assumption of water solid been assessed for fire scenarios (e.g., flow diversion, LOCA outside containment, etc).*

c. *Please discuss which Intersystem LOCA (ISLOCA) paths were considered for the FPRA and if multiple spurious operations (MSOs) impacted any of the valves in these paths. Please describe impacts to the FPRA from the consideration of MSOs on these paths and their significance to the FPRA results*.

Waterford 3 Response

a. Flow Diversion Paths for LERF

Two types of LERF scenario could be affected by flow diversions: containment isolation failure and Intersystem LOCA (ISLOCA). Both of these LERF scenarios involve release paths that bypass the containment. Since the release paths are via piping systems that penetrate containment, flow diversions either inside or outside containment could potentially affect containment integrity and contribute to large early release (LER). The LERF model bounds the contribution of potential flow diversions inside or outside containment in systems that penetrate containment, as explained in the following.

(1) Containment Isolation Failure. Containment Isolation failure is explicitly included in the LERF model, via fault tree logic. The assumption in the LERF model is that if Containment Isolation fails before or after core damage, LER occurs. No credit is taken for potential mitigation of the release by closed systems or water outside containment; therefore, flow diversions outside containment, which could compromise the mitigating effects of piping, are bounded by the assumption of no mitigation outside containment, and do not need to be included in the LERF model. For flow diversions inside containment, any flow diversion would need to pass through a containment penetration in order to contribute to LER; the modeling of Containment Isolation ensures that these potential flow diversions are accounted for in LERF. In other words, a flow diversion for which Containment Isolation is intact would not be a LER; conversely, a containment isolation failure in a penetration open to the containment atmosphere would be a LER in the LERF model, whether or not there was a flow diversion inside containment. Containment Isolation failures for penetrations associated with systems connected to the RCS is explicitly modeled in the ISLOCA modeling, discussed next.

(2) ISLOCA. In ISLOCA, isolation valves between high and low pressure portions of piping systems connected to the RCS fail, leading to a LOCA outside containment and core damage, with the containment bypassed. Since an ISLOCA scenario includes failure of Containment Isolation valves in the piping systems affected, they are similar in this respect to the Containment Isolation failure above, but in the Internal Events and Fire PRA models are treated as separate scenarios. ISLOCA is characterized by the high pressure-low pressure interface, with the high pressure side connected to the RCS and the low pressure side connected to piping outside containment. In the ISLOCA model, it is assumed that piping and valves outside containment, beyond the isolation valves includes in the ISLOCA model, do not provide any mitigating function. Therefore, if a failure of the isolation valves modeled in the ISLOCA model occurs, it is assumed that a containment bypass exists. If core damage results from the ISLOCA, it is assumed that a LEF occurs. Any failures inside containment that contribute to the ISLOCA, including flow diversions, are included in the ISLOCA modeling. Flow diversions outside containment are not relevant to LERF because the piping systems outside containment are already assumed failed (i.e., not credited for ISLOCA mitigation). Therefore, the

ISLOCA modeling bounds the effects of flow diversions on the LERF for ISLOCA scenarios.

b. LERF Screening for Water Solid Paths

PRA-W3-05-002, Appendix C, Table C-1, describes screening criterion "b – Water Solid" as "Water solid penetrations are torturous paths and are insignificant with respect to consequences". This LERF screening criterion (b) was used to screen out several groups of containment penetrations from needing their associated components added to the FPRA component list. The application of screening criterion b to the containment penetrations did not actually rely on the "water solid" nature of the penetrations, but rather on the fact that the piping systems associated with the penetrations were closed inside containment. The specifics of the application of screening criterion b to screened penetrations are:

(1) Main steam, feedwater, and blowdown lines are closed inside containment; although they are connected to the steam generator secondary, the steam generator tubes separate the secondary from the primary (RCS)—since a steam generator tube rupture is not fire-induced, there is not a containment bypass scenario for the main steam, feedwater, and blowdown lines.

(2) CCW to the containment fan coolers is contained in closed loops inside containment, so they are not connected to the containment atmosphere or RCS and thus cannot produce a containment bypass without a CCW piping failure inside containment, unrelated to a fire.

(3) CCW to control element drive mechanism coolers and reactor coolant pump (RCP) seal cooling are closed systems inside containment and are not directly connected to the RCS. The CCW flow to the RCPs for cooling controlled bleedoff of RCS water through the seals is contained in a closed heat exchanger baffle assembly between the RCP impeller and the shaft seals and does not contact RCS water.

(4) For the Safety Injection Tank recirculation header to RWSP, there is a closed manual valve (SI-3433) and a locked closed manual valve (SI-344), which are not susceptible to fire, to prevent containment bypass; the application of screening criterion b is secondary—the line is closed inside containment upstream of the credited isolation valves.

(5) Containment pressure wide range instrumentation lines are closed both inside and outside containment.

(6) Fire suppression to RCPs[1] is a deluge system (sprinkler heads are open) with a check valve inside containment and a normally open containment isolation valve outside containment (FP-601A and FP-601B, for deluge systems 1 and 2); upstream of the containment isolation valve in each line is another check valve and a normally closed deluge valve, behind which (upstream) is water filled (and closed) piping. These fire suppression lines also supply water to the Airborne Radioactivity Removal System E-13 charcoal filter units, which are protected by the same valves as the RCP deluge lines (the same supply header through containment penetration 60 supplies the RCP deluge lines and the E-13 charcoal filter deluge lines). Therefore, the bypass path through the FP lines to the RCPs and E-13 filters is prevented by two check valves, a closed deluge valve, and closed, water filled piping.

[1] Although fire suppression to the RCPs is no longer credited in the Fire Protection design basis, these lines are still present in containment. The FP-601 containment isolation valves are normally open because the system supplies fire suppression water to the E-13 charcoal units.

c. ISLOCA Paths and MSOs

The ISLOCA paths in the FPRA are those modeled in the internal events PRA. The following screening criteria were applied in selecting potential ISLOCA paths; these criteria were applied to systems passing through containment penetrations:

(1) The fluid system must connect to the RCS high-pressure system considering possible valve positions;

(2) the fluid system pathway must have the capability to produce a high-to-low-pressure interface;

(3) the fluid system must have the capability to be over-pressurized at power;

(4) the pipe diameter must be greater than 1” for water, 2" for steam;

(5) the fluid system has fewer than 4 isolation valves, or if there are 4 or more isolation valves, they must be normally closed, or if they are check valves, they must receive periodic testing.

If these criteria were satisfied, the pathway was considered susceptible to ISLOCA and was included in the internal events PRA. Two types of pathways (for a total of 6 piping penetrations/lines) were found to be susceptible to ISLOCA: Shutdown Cooling (SDC) suction lines (2 lines) and Low Pressure Safety Injection (LPSI) injection lines (4 lines). These six lines were thus included in the FPRA model. Since the ISLOCA modeling in the FPRA included the valves in each line that must fail in order for an ISLOCA to occur, and with the appropriate failure mode (generally failing open), the effect of MSOs on these ISLOCA paths is modeled. Each of the valves in these lines is linked to the associated cables via the cable-to-component database in the FPRA; any fire that affects multiple cables associated with an ISLOCA path is modeled in the FPRA via this cable-to-component linkage.

Several types of pathway (High Pressure Safety Injection, Hot Leg Injection, and Safety Injection Tank drains) were screened out from inclusion in the ISLOCA model because there were at least 4 isolation valves. In each of these lines, there are at least 2 check valves preventing ISLOCA; since check valves are not susceptible to fire, these check valves provide protection against ISOCA due to potential MSOs affecting the motor-operated or air-operated valves in the flow paths.

Of the two types of ISLOCA pathways included in the FPRA, only the SDC suction line pathway is significant. The LPSI injection pathways include 2 check valves in each line, so fire-induced ISLOCA on these lines is very unlikely and their contribution to CDF is negligible. ISLOCA risk is dominated by the SDC suction lines, but even these pathways are very unlikely, with a Fussel-Vesely (fractional contribution to CDF) of 2E-6. The dominant ISLOCA scenarios are on the 2 SDC suction lines and is a fire-induced spurious opening of the downstream RCS isolation valve (SI-405A or B, which are air-operated valves) with a fire-induced proper polarity 3 phase hot short with the deenergized motor-operated upstream RCS isolation valve (SI-401A or B) in the same line. A 5E-8 proper polarity 3 phase hot short probability of 5E-8, based on NUREG/CR-6850, Chapter 9, Section 9.5.2.2(2), is used in this ISLOCA calculation.

In summary, potentially significant ISLOCA pathways are modeled in the FPRA, and include the effects of MSOs; the resulting fire-induced ISLOCA scenarios are not significant contributors to fire risk because of either the presence of at least 2 check valves in each line preventing ISLOCA (in the case of LPSI) or a deenergized power cable to a motor-operated isolation valve in each line (in the case of the SDC suction).

RAI PRA 56

Scenario Modeling

There are only two control room abandonment scenarios in Table W-1. Please describe the reasonableness review of the low CDF and LERF contribution from control room abandonment scenarios, and the findings. Additionally, please discuss whether potential fire impacts on alternate shutdown capability were considered for applicable fire areas for these scenarios, and how the reasonableness review considered them.

Waterford 3 Response

Total CDF contribution for all control room abandonment scenarios was just above 1E-6 in the LAR submittal. This was approximately 3% of the total Fire PRA CDF. This was considered reasonable given the relatively low control room abandonment probability due to the Waterford 3 control room configuration.

However, the aggregate effect of the conclusions drawn during the efforts to support RAI responses has led to the desire to adjust some analyses to more fully comply with approved NUREG/CR-6850 methodologies. The MCR abandonment analysis is one example of this. This analysis is currently being revised. Updated results for control room abandonment scenarios will be presented with RAI PRA 40.

RAI PRA 57

VFDRs

a. *Many VFDRs have a statement" ... variance has no corresponding PRA basic event and by definition has insignificant risk.» Please clarify this statement.*
b. *VFDR 2-19 and 2-26. 4S0V SWITCHGEAR BUS 3A31-S SSD-ESWGR-31A. Potential fires for these VFDRs could affect non-SSD equipment (e.g., shield building ventilation fans) according to the descriptions. Please explain why the VFDR descriptions state that these VFDRs could affect Vital Auxiliaries (Electrical) Performance Criteria.*
c. *It appears that train B systems are the success path for fires in RAB 7 A from Table B-3.*
    I. *If train B is to be the fire-protected train for SSD, please describe why there are VFDRs for both Charging Pumps A and B (7A-OS and 7 A-12 respectively) in the fire area.*
    II. *ii. Please describe whether a VFDR issue (e.g., separation) between charging pump trains is in this fire area. If so, describe why there are two VFDRs and not only one VFDR for that condition. Please explain the success path for this fire area if a fire scenario impacts both charging pumps.*

Waterford 3 Response

a. In identification of VFDRs, Waterford used more rigorous criteria than was necessary in meeting the guidance requirements. Equipment and cables used to support SSA recovery actions were identified with VFDRs. This led to numerous VFDRs that had no corresponding PRA basic event. It also led to several items used to support actions related to cold shutdown (and beyond the scope of the PRA model). VFDRs with the

text listed in the RAI are ones associated with equipment not included in the PRA model and have no quantifiable risk relevance.

b. The load associated with this breaker is not required for safe shutdown in this area. Fire damage to this cable may prevent the breaker from providing fault overcurrent trip capability for the power cable in the area. The statements in the B-3 table are in error. VFDRs 2-19 and 2-26 can not impact Vital Auxiliaries (Electrical) Performance Criteria.

c. In the deterministic evaluations for RAB-7A failures were identified as;

   i) This is an example of over rigorous VFDR classification. There is a separation issue in the area and fires impacting both A and B charging pumps are possible. The train B pump VFDR is based solely on a credited SSA recovery action (one that is not included in LAR attachment G).

   ii) This is an example of over rigorous VFDR classification. There is a separation issue related to Fire Area 7A for Charging pumps 1A and 1B. The FRE for this area accounts for the failure of both pumps in the delta risk evaluation.

RAI PRA 58

Fire Risk Evaluations (FREs)

a. Please describe the "TRUE" method used for the FREs, and clarify to which case the "TRUE" value is applied.

b. For those VFDRs with a status of "open" in the LAR, please discuss their dispositions as well as associated FRE (or other assessment) conclusions, and identify any associated commitments or implementation items. If a FRE had been performed for the VFDR, provide the risk associated with the VFDR for the fire area (ΔCDF and ΔLERF).

Waterford 3 Response

a. The TRUE method does not refer to compliance with NFPA 805.

   The "TRUE" method refers to a calculation method whereby the basic event/s (representing a component/s to be failed) is/are set to "True" in the model. The model is then compressed. This allows the model to act as if the component doesn't exist (i.e.,it is a failed component). The model is then quantified, giving the risk (e.g. CDF) of the plant in that particular arrangement.

   The non-compliant case includes basic events that no credit can be given to when compliant with NFPA 805. These basic events are items such as modifications to the plant or operator action that reduce the risk of fire leading to an undesirable state.

b. Every VFDR entry in the B-3 table with ‘open’ status is associated with a modification. In every open case the B-3 disposition identifies an Attachment S implementation item. For all of these cases, the risk measurement for the corresponding VFDR for the fire area (ΔCDF and ΔLERF) is provided in the FRE calculations identified for each fire area.

RAI PRA 59

Recovery Actions

a. *Step 3 in Attachment G of the LAR states that all of the operator manual actions and RAs were reviewed for adverse impact, and that none of the actions were found to have an adverse impact on the FPRA. Please explain how this conclusion was arrived at for each RA.*

b. *Please describe how fire scenarios in the FPRA have considered the missing or de-rated fire dampers and their impact on affected fire areas and operator actions/RAs.*

c. *Please describe whether fire scenario sequences involve more than one RA (i.e., are there combinations of RAs in a sequence). If so, describe whether this has been reviewed to ensure the combined RA probabilities are reasonable.*

d. *Table G-1 of the LAR shows two RAs for RAB 6 for EFW valves. During the audit, it was noted in discussions that these RAs support meeting risk acceptance criteria for the fire area. Please review of these RAs was not able to conclude the HRA quantification was conservative. Perform a sensitivity analysis which reflects a conservative modeling of these RAs. Please explain the sensitivity analysis inputs, and their impact relative to meeting the risk acceptance criteria.*

e. *Please describe whether or not the RAs in Table G of the LAR are new. Describe whether the risk of previously approved RAs as well as new RAs has been included in the fire area delta risk evaluations.*

Waterford 3 Response

a. Adverse operator actions were considered based on an action that would remove a safety function (or challenge a PRA modeled non-safety function) from a component that may or may not be damaged. An example of an adverse operator action is the following action in RAB 5

   1. Stop the running Swgr Main Ventilation Air Handling Unit AH-25:

      - SVS-0001A

        or

      - SVS-0001B

   This action to stop switchgear HVAC to prevent soot accumulation on the filter is only needed for the worst case fires. Otherwise, HVAC is needed to prevent switchgear failure on high temperature.

   Each RA listed in the LAR was examined to determine if the action could adversely impact a safety function or other PRA credited function.

c. The following is an excerpt from PRA-W3-05-001 section 2.2.2

   "As part of the Fire PRA walkdowns conducted for the purpose of fixed ignition source notification (NUREG/CR 6850 Task 6) and for scoping scenario development (NUREG/CR 6850 Task 8), the PAU enclosing walls, floors, ceilings, doors and other openings [including fire dampers] in the same were observed. Defects in construction or current degraded conditions that would have rendered the feature unsuitable for reliance upon as a barrier were noted as exceptions."

Therefore, as a part of this walkdown the omission of barriers would have been addressed in the potential for a boundary to contain a fire and would be reflected in the zone definition. This would be mapped to the FPRA model and would be reflected in the FRE assessment as a multiple compartment assessment. Examination of the FPRA FRANC input in comparison to the defined VFDRs did not identify any components or changes necessary to address the omitted dampers. Therefore, the assumption is made that the barrier was addressed within the scope of the plant partitioning notebook and that the FREs adequately reflect the impact on risk.

c. Combinations of operator actions are addressed in the FPRA quantification. Dependence between the operator actions is evaluated using the HRA methodology for dependence. The higher HEP for the fire-induced HRA is also translated to HRA combinations. The HRA combinations are evaluated in the internal events model for consistency with the CDF and LERF scenario. The HRA combinations for fire were compared to its internal events counterpart and new fire-related HRA combinations were compared with the fire HRA combinations for consistency.

d. The actions listed in LAR Table G-1 for RAB 6 are not required to meet risk acceptance criteria for the fire area. A review of the RAB 6 FRE found that valves EFW-228A and EFW-229B accounted for the majority of the delta risk. Recovery actions to ensure these valves remain open were included in RAB 6. These actions were included due to risk, but were not explicitly required to meet risk acceptance criteria. The fire risk evaluation for RAB 6 shows that the fire area is below the risk acceptance criteria without crediting these actions. No sensitivity on these actions is needed. The risk associated with these actions is bounded by the delta risk value of the FRE.

e. No new recovery actions were identified for transition to NFPA 805. All actions listed in LAR attachment G are also included in OP-901-524 or OP-901-502. The risk associated with all RAs is bounded by the FRE delta risk values. A VFDR was identified for each RA in each relevant fire area.

SAFE SHUTDOWN ANALYSIS (SSA)

RAI SSA 01

Section 2.4.2 of NFPA 805 defines the methodology and requirements for performing a Nuclear Safety Capability Assessment (NSCA).

LAR Table B-2 references Waterford calculation EC-FOO-026, "Post-Fire Safe Shutdown Analysis," Revision 2, for many of the Table B-2 elements associated with the NSCA requirements; however, this calculation is the Appendix R/BTP CMEB 9.5.1 safe shutdown analysis and does not adequately address all elements necessary to meet NFPA 805 Section 2.4.2. Please provide the following:

a. *A description of the analyses, databases, and documentation that constitute the NSCA, including how the current Appendix R safe shutdown analysis and other documentation (e.g., the MSO analysis, the PRA, and the NPO analysis) will be transitioned and integrated to demonstrate compliance with the NSCA requirements of Section 2.4.2 (and associated subsections 2.4.2.1-2.4.2.4) of NFPA 805.*

b. *Please describe how the above documents are integrated in the plant equipment database, models, and engineering tools for maintaining the nuclear safety performance criteria post-transition.*

Waterford 3 Response

a) Engineering packages for MSO (EC19062) and NPO (EC15964) are complete and revision 3 of the SSA (ECF00-026) contains this information to include the NFPA-805 (Section 2.4.2 and associated subsections 2.4.2.1-2.4.2.4) criteria and analysis. Updating of the SSA is occurring as each engineering package on each subject is generated.

   The Fire Safe Shutdown Analysis does not contain the additional equipment, components and cable utilized by the PRA analysis (NFPA-805, section 2.4.3). As the PRA analysis utilizes different acceptance criteria, the additional equipment added by PRA was not needed for the safe shutdown analysis. The PRA contains the equipment and cables used by the safe shutdown analysis plus additional equipment and cable needed for the PRA analysis. See part b) of this response for additional detail.

b) Prior to the transition to NFPA-805, calculation EC-F00-026 will be updated to reflect all the NSCA criteria. The recovery actions and fire wrap presently in the document will be updated with the VFDR information from the Fire Risk Evaluations (FRE).

   The Plant Data Management System (PDMS) maintains all the routings for the cables utilized in the SSA and the PRA analysis.

   During this transition process Waterford 3 will also be transitioning the Fire protection analysis to utilize ARC software. This software will incorporate the NSCA information and the Fire Risk analysis information to produce the NFPA-805 integrated analysis. This software will access the Plant Data Management System (PDMS) database, which routes all the cables utilized by both approaches by raceway and by fire area in the Plant.

LAR Attachment S, item S2-13, implements the actions above.

RAI SSA 02

LAR Section 4.2.1.1 and Table B-2 are based on NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," Revision 1, January 200S (ADAMS Accession No. MLOS031029S). Regulatory Guide (RG) 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, cites NEI 00-01, Revision 2, May 2009 (ADAMS Accession No. ML09177026S), as the acceptable method for circuit analysis.

Please provide a gap analysis on the differences between the alignment of the Waterford methodology with NEI 00-01, Revision 1, and NEI 00-01, Revision 2, cited in RG 1.205. As part of this gap analysis, clarify if actions are necessary to manually operate valves post-fire that are located in the fire area of concern and may have been exposed to the fire (refer to NEJ 00-01, Revision 2, Section 3.2.1.2 example discussion for rising stem valves). Please discuss the impact of fire exposure on manual operation of the valves.

Waterford 3 Response

A review of NEI 00-01, Section 3.0, Rev. 2 vs Rev. 1 provides the following gaps that will require the Safe Shutdown Analysis (NSCA) to be updated:

- Rising stem valves that are located in the fire area of concern and may need to be manually operated post fire will be evaluated for operability. Preliminary review indicates no rising stem valves in the affected fire area will need to be manually operated.
- The required for hot shutdown / important for safe shutdown guidance will be incorporated into the NSCA. Currently, Waterford treats everything as “required”.
- IN92-18 valve assessments are complete with final resolution listed in Attachment S, items S1-1 and S1-2.
- The new criteria introduced in section 3.5.1.1 for evaluating inter-cable hot shorts for proper polarity DC circuits will be need to be incorporated into the NSCA. (See SSA RAI-05)
- Circuit breaker coordination calculations will be reviewed to ensure that the section 3.5.2.4 criteria for breakers that have internal breaker tripping devices and do not require control power to trip the breaker, assure that the time-current characteristic curve for any affected load breaker is to the left of the time-current characteristic curve for the bus feeder breaker and that the available short circuit current for each affected breaker is to the right of the time-current characteristic curve for the bus feeder breaker or that the bus feeder breaker has a longer time delay in the breaker instantaneous range than the load breaker. For breakers requiring control power for the breaker to trip, the availability of the required control power must be demonstrated in addition to the proper alignment of the time-current characteristic curves.
- PRA utilized revision 1 of NEI 00-01 for their analysis. Preliminary review indicates no impact on the PRA from revision 2 criteria.

In summary, the Safe Shutdown Analysis (NSCA) and the PRA will be updated to include the additional requirements introduced by revision 2 of NEI 00-01 prior to implementation of NFPA-805. An implementation item has been added to Attachment S to cover the updating of the NSCA to cover the transition to revision 2, section 3.0 of NEI 00-01.

RAI SSA 03

Table B-2, element 3.3.3.1 contains the following statement in the Alignment Basis:

> *For some electrically supervised safe shutdown earthquake (SSE) circuit analysis was not completed. These were for device in air instrument support was not credited and loss of air position or was not the same as loss of power, power supplied was not diesel backed and it was determined not to credit non-diesel backed power supplies in order to minimize component selection.*

This paragraph is fractured and incomplete. Please provide the corrected text for this alignment basis statement.

Waterford 3 Response

Since the submittal of the LAR, the Instrument Air compressors have been added to the SSEL and the circuits routed. Thus, this fragmented sentence no longer applies and will be removed from the B-2 Table.

RAI SSA 04

For several of the entries in Table 8-2, the Reference Documents include references to RAIs for the Harris Nuclear Plant (HNP) and Oconee Nuclear Station (ONS). There is no discussion in the alignment basis statements regarding the applicability or similarity of these RAIs to Waterford. Please provide additional clarification or discussion of the applicability of these pilot-plant RAIs to the alignment of the Waterford NSCA with the associated provisions of NEI 00 01, Revision 2.

Waterford 3 Response

When pilot plant RAI's are referenced, it was only to indicate that the question is considered in the Waterford response, so these references can be ignored.

The B-2 table will be updated to remove these references.

RAI SSA 05

NEI 00-01, Revision 2, Section 3.5.1, requires consideration of proper-polarity hot shorts in certain dc control circuits for non-high-low pressure interface components. NEI 00-01, Revision 1 (referenced in the LAR), Section 3.5.1.5.D also includes criteria for consideration of proper-polarity dc shorts. LAR Table 8-2, elements 3.5.1 and 3.5.2, state that dc proper polarity shorts are not considered except for high-low pressure interface components. Please provide the basis

and justification for not considering proper-polarity dc shorts in non-high-low pressure interfacing components as described in NEI 00-01, Revision 2.

Waterford 3 Response

- NEI 00-01R1, 3.5.1.5 D. For cases involving direct current (DC) circuits, the potential spurious operation due to failures of the associated control cables (even if the spurious operation requires two concurrent hot shorts of the proper polarity, e.g., plus-to-plus and minus-to-minus) should be considered when the required source and target conductors are each located within the same multiconductor cable.
- NEI 00-01R2, 3.5.1.1. **Circuits for "required for hot shutdown" components:** Because Appendix R Section III.G.1 requires that the hot shutdown capability remain "free of fire damage", there is no limit on the number of concurrent/simultaneous fire-induced circuit failures that must be considered for circuits for components "required for hot shutdown: located within the same fire area. For components classified as "required for hot shutdown", there is no limit on the duration of the hot short. It must be assumed to exist until an action is taken to mitigate its effects. Circuits required for the operation of or that can cause the mal-operation of "required for hot shutdown" components that are impacted by a fire are considered to render the component unavailable for performing its hot shutdown function unless these circuits are properly protected as described in the next sentence. The required circuits for any "required for hot shutdown" component, if located within the same fire area where they are credited for achieving hot shutdown, must be protected in accordance with one of the requirements of Appendix R Section III.G.2 or plant specific license conditions.
- The Safe Shutdown Analysis (NSCA) will be updated to indicate compliance with the proper polarity DC shorts per revision 2 of NEI 00-01 prior to implementation of NFPA-805. A work task item will be added to Attachment S to cover the updating of the NSCA (See RAI SSA-02).

RAI SSA 06

Attachment G of the LAR describes the process for identifying RAs necessary to meet the requirements of NFPA 805. The results of Step 4 state that six control room evacuation RAs were identified that require additional feasibility assessment. Please confirm that the additional feasibility assessment is limited to the availability of emergency lighting to be installed as described in Attachment S, Plant Modification Item S1-6.

Waterford 3 Response

The results of step #4 section of Attachment G states: "Six control room evacuation actions were identified that require additional feasibility assessment. A plant modification is needed to confirm compliance with the feasibility requirements (HVAC local control station actions and associated emergency lights for a control room fire). This modification is provided in Attachment S." The specific equipment is:

- AH-13B, RAB HVAC Equipment Room AHU
- AH-10B, CCW Pump Room B AHU
- AH-2B, Safeguards Pump Room B AHU
- AH-2D, Safeguards Pump Room B AHU

- AH-17B, EFW Pump Room B AHU
- AH-18B, Charging Pump Room B AHU

Item S1-6 in Attachment S is provided for this modification.

The present understanding is that the additional emergency lighting listed above is the only additional scope of work needed to comply with the feasibility assessment. However, when the Engineering Change (EC) for this task is finalized (Attachment S, Line Item S1-6) for these new manual actions, EC-F00-026 Appendix E (Manual Action Feasibility Study) and FAQ-07-0030 Revision 5 (11 feasibility criteria) will again be reviewed to verify that the manual actions are feasible. This will be performed through existing processes and procedures including EN-DC-115 (Engineering Changes Process) and EN-DC-128 (Fire Protection Impact Reviews). If additional work (such as training and procedural updates) is needed to comply with the feasibility requirements it will be addressed in this new EC.

RAI SSA 07 (to be addressed in the 90 Day Response)

RAI SSA 08

Attachment G states the Manual Action Feasibility Study in Appendix E to Calculation EC-FOO-026, "is generally consistent" with the approach in FAQ 07-0030, Revision 5. Appendix E also describes Appendix R manual actions.

a. *Please provide additional discussion with regard to the alignment of the approach in the study with the 11 feasibility criteria in the FAQ.*
b. *Please confirm that the actions in Appendix E include recovery actions to meet NFPA 805 or provide discussion of the feasibility analysis performed.*

Waterford 3 Response

a) Waterford 3 Alignment with *FAQ 07-0030, Table B-TBD, Feasibility Criteria – Recovery Actions:*

1 Demonstrations:
The proposed recovery actions should be verified in the field to ensure the action can be physically performed under the conditions expected during and after the fire event.

SSA Appendix E, Section 5.1.1: Accessibility:

a. The component to be operated is physically accessible (e.g., the component is within reach without a ladder, unless the ladder is staged).
b. The manual action can be performed by the operator(s) (e.g., opening an MOV is not feasible if the valve has no handwheel operator).
c. Actions requiring entry into or travel through the affected Fire Area have been evaluated successfully based on factors such as combustible loading, location of combustibles, availability of detection and suppression equipment, and the physical

properties of the component. Entry into the affected Fire Area is deemed acceptable after 1 hour without further evaluation.

Timed recovery actions are walked down, and placed in the operating procedure, by operations to indicate the demonstrated time to perform the action. (OP-901-502, OP-901-503 & OP-901-524)

2 Systems and Indications:
Consider availability of systems and indications essential to perform the recovery action.

SSA Section 6.4.6 “Plant Monitoring Instrumentation”, (in the context of post-fire safe shutdown operation) consists of those instruments or local gauges/indicators that are necessary to monitor the operation of primary safe shutdown system parameters and the operation of those systems or components that provide required support functions. The instrumentation selected is based on the guidance of NRC Information Notice 84-09, which identifies the minimum monitoring capability considered necessary for a PWR

3 Communications:
The communications system should be evaluated to determine the availability of communication, where required for coordination of recovery actions.

SSA Appendix E, Section 6.12: A generic assessment of communications necessary to support safe shutdown was performed. This review considered items such as reliability and redundancy of plant communications systems following a fire and loss of offsite power. Items such as the use of portable two-way radios, are reviewed to ensure that:

1. an adequate number of portable radios are available to support worst-case fire scenarios,
2. multiple organizations (fire brigade, security, etc.) are not relying on the same pieces of equipment, and
3. The equipment is properly tested and periodically inventoried.

4 Emergency Lighting:
The lighting (fixed and/or portable) should be evaluated to ensure sufficient lighting is available to perform the intended action.

SSA Appendix E, Section 7.1.3: Emergency lighting is identified in the Appendix R Database and on the Lighting Drawings. Based on a review of the drawings, sufficient 8 hour battery backed emergency lighting is installed to support access/egress to local equipment for required hot standby manual actions. Special Tests (STI-2002-0003-01, Revision 1) were performed with normal lighting secured to ensue the adequacy of the emergency lighting.

5 Tools-Equipment:
Any tools, equipment, or keys required for the action should be available and accessible. This includes consideration of SCBA and personal protective equipment if required. (This includes staged equipment for repairs).

SSA Appendix E, Section 7.1.6: The credited repairs involve the replacement of blown fuses at various locations, but primarily in the Switchgear Areas (RAB8A, RAB8B, and RAB8C). These fuse replacements are credited to achieve cold shutdown only, and only for the Control Room / Cable Vault scenario. Dedicated fuses are staged for use.

6 Procedures:
Written procedures should be provided.

SSA Appendix E, Section 7.1.2: Manual actions are addressed and identified by the following Operating Procedures;
OP-901-502, “Evacuation of the Control Room and Subsequent Plant Shutdown”,
OP-901-503, “Isolation Panel Room Fire” and
OP-901-524, “Fire in Areas Affecting Safety Shutdown”.

7 Staffing:
Walk-through of operations guidance (modified, as necessary, based on the analysis) should be conducted to determine if adequate resources are available to perform the potential recovery actions within the time constraints (before an unrecoverable condition is reached), based on the minimum shift staffing. The use of essential personnel to perform actions should not interfere with any collateral industrial fire brigade or control room duties.

SSA Appendix E, Section 5.1.4: Adequate time and manpower are available to perform the action such that the plant is not placed in an unrecoverable condition.

SSA Appendix E, Section 6.9: Time and manpower are assessed for the population of manual operator actions for each Fire Area. The adequacy of time and manpower considers the minimum Control Room and plant staffing, with considerations made for fire brigade response.

8 Actions in the Fire Area:
When recovery actions are necessary in the fire area under consideration or require traversing through the fire area under consideration, the analysis should demonstrate that the area is tenable and that fire or fire suppressant damage will not prevent the recovery action from being performed.

SSA Appendix E, Section 5.1.1.c: Actions requiring entry into or travel through the affected Fire Area have been evaluated successfully based on factors such as combustible loading, location of combustibles, availability of detection and suppression equipment, and the physical properties of the component.

SSA Appendix E, “Manual Action Feasibility Study”, Section 7.2: The following areas specify hot standby actions requiring access through or to the affected area of concern.

9 Time:
Sufficient time to travel to each action location and perform the action should exist. The action should be capable of being identified and performed in the time required to support the associated shutdown function(s) such that an unrecoverable condition does not occur. Previous action locations should be considered when sequential actions are required.

SSA Appendix E, Section 5.1.4: Adequate time and manpower are available to perform the action such that the plant is not placed in an unrecoverable condition.

SSA Appendix E, Section 6.9: The Functional Requirements Timeline Analysis addresses best estimate time frames in which manual operator actions should occur to restore post-fire safe shutdown functions so the plant is not placed in an unrecoverable condition. The time sensitivity of all manual operator actions for each Fire Area is reviewed to assess the adequacy of time and manpower.

10 Training:
Training should be provided on the post-fire procedures and implementation of the recovery actions.

Training is conducted in accordance with the Entergy Training and Qualification Procedures and the Waterford Operations Training Manual (TM-OP-100). Control Room/Cable Vault evacuation (OP-901-502) is currently performed every 2 years and the rest of the plant (OP-901-503 and OP-901-524) is performed every 4 years. The Training Department is designated as affected on a change document which modifies an operating procedure. (EN-DC-115).

11 Drills:
Periodic drills that simulate the conditions to the extent practical (e.g., communications between the control room and field actions, the use of SCBAs if credited, the appropriate use of operator aids).

Drills were not specifically called out for the recovery actions, thus Attachment S (Item S2-17) was generated to incorporate this action.

b) The actions covered in Appendix E include recovery actions to meet NFPA 805 Safe and Stable Hot Shutdown.

RAI SSA 09

LAR, Table B-3, Fire Areas, RCB, Roof E and Ware described as performance-based areas per NFPA 805, Section 4.2.4.2. Please clarify the "DID Maintained" part of the fire risk summary which states that these areas are compliant with deterministic requirements of NFPA 805, Section 4.2.3.

Waterford 3 Response

The RCB, Roof E and Roof W are Performance-based fire areas.

The “DID Maintained” section of the B-3 Table will be corrected.

RAI SSA 10

LAR, Attachment D, describes the methodology for evaluating non-power modes of operation. The discussion in Attachment D and the reference calculation (ECF09-005, "NFPA 805

Transition, Non-Power Operating Mode") indicate that RAs are credited for restoring key safety functions. Please describe the action and confirm that the action has been reviewed for feasibility, when a RA(s) is the only means to restore or recover the key safety function.

Waterford 3 Response

Waterford has not determined that recovery actions will be used to restore KSFs during Higher Risk Evolutions. All of the options described by FAQ 07-0040, including controlling or prohibiting combustible storage, controlling or prohibiting hot work, rescheduling work to periods of lower risk and/or post firewatches may be employed during higher risk operating states to manage fire risk during non-power operations. Should a recovery action be utilized as a means of reducing fire risk during a NPO High Risk Evolution (HRE), then the action would be evaluated for feasibility. This feasibility evaluation is not required to be in accordance with FAQ 07-0030, but would ensure the equipment is functional and that operators are available to perform the action within the time frame required. Calculation ECF-09-005 will be revised to emphasize this approach for managing fire risk during non-power Operational periods. LAR Attachment S, item S2-13, implements the above actions.

RAI SSA 11

LAR, Attachment D, states that scoping fire modeling per NUREG/CR-6850 was performed to determine locations where pinch-points could be eliminated. Provide additional discussion of how the fire modeling was performed and documented and describe how the model assumptions will be incorporated in plant procedures and that the basis for eliminating these pinch points are maintained in the plant post-transition.

Waterford 3 Response

The NPO analysis, which utilized scoping fire modeling results per NUREG/CR 6850, determined a zone of influence (ZOI) from prescribed fire sources (in-situ ignition sources and transient combustibles) and was used to evaluate potential damage to equipment and components (mostly cable) from a fire event. The ZOI is defined for installed equipment ignition sources and various transient combustibles sources in NUREG/CR-6850. A compilation of these ignition sources and associated ZOI's were tabulated.

The NPO analysis, which includes the scoping fire modeling, will affect the Outage Planning Procedures (this would include PLG-009-014, "Conduct of Planned Outages"; See Attachment S, Item S2-7). The NPO analysis may also add "Level 1" transient combustible control areas in the plant per EN-DC-161. The new level 1 areas will not permit unattended transient combustible in all or a portion of these areas.

The NPO design documents that establish the basis for the procedures and identified pinch points are governed by the plant configuration control process, which includes the use of approved procedures (i.e. EN-DC-115, EN-DC-126, EN-DC-128, EN-LI-100, etc) and/or processes.

FIRE MODELING (FM)

MONITORING PROGRAM (MP)

RAI MP-01

NFPA 805, section 2.6 "Monitoring" states that "a monitoring program shall be established to ensure that the availability and reliability of the fire protection systems and features are maintained and to assess the performance of the fire protection program in meeting the performance criteria" and that "Monitoring shall ensure that the assumptions in the engineering analysis remain valid."

Specifically, NFPA 805, Section 2.6 states that

> *(2.6.1) "Acceptable levels of availability, reliability, and performance shall be established. "*
>
> *(2.6.2) "Methods to monitor availability, reliability, and performance shall be established. The methods shall consider the plant operating experience and industry operating experience."*
>
> *(2.6.3) "If the established levels of availability, reliability, or performance are not met, appropriate corrective actions to return to the established levels shall be implemented. Monitoring shall be continued to ensure that the corrective actions are effective."*

Section 4.6, "Monitoring Program" of the Transition Report states that the NFPA 805 monitoring program will be implemented "after the safety evaluation issuance as part of the fire protection program transition to NFPA 805" (Table S-3, Implementation Items, item 11-805-089 of the Transition Report).

Furthermore, the licensee has committed to comply with FAQ 10-0059. The NRC staff noted that the information provided in Section 4.6, "Monitoring Program" of the Transition Report is insufficient for the staff to complete its review of the monitoring program, and as such, is requesting that the following additional information be provided:

a. *A description of the process by which systems, structures, and components (SSCs) will be identified for inclusion in the NFPA 805 monitoring program, including the approach to be applied to any fire protection SSCs that are already included within the scope of the Maintenance Rule program.*
b. *A description of the process that will be used to assign availability, reliability, and performance goals to SSCs within the scope of the monitoring program including the approach to be applied to any SSCs for which availability, reliability, and performance goals are not readily quantified.*
c. *A demonstration of how the monitoring program will address response to programmatic elements that fail to meet performance goals (example: discrepancies identified in programmatic areas such as combustible controls program).*
d. *A description of how the monitoring program will address fundamental fire protection program elements.*
e. *A description of how the guidance in EPRI Technical Report 1006756, "Fire Protection Equipment Surveillance Optimization and Maintenance Guide" if used, will be integrated into the monitoring program.*
f. *A description of how periodic assessments of the monitoring program will be performed taking into account, where practical, industry wide operating experience including whether this process will include both internal and external assessments and the frequency at which these assessments will be performed.*

g. *Confirmation that periodic NFPA 805 assessments (audits) of the fire protection program will be conducted under the existing Fire Protection Quality Assurance Program. If not, please describe the process that will be used to conduct these assessments.*

Waterford 3 Response

Waterford will use the process outlined in NEI 04-02 and FAQ 10-0059 to develop an NFPA 805 Monitoring Program. Specific responses to this RAI are summarized below.

a. The process by which SSCs are identified (Phase 1 – "Scoping", and Phase 2 – "Screening") are identified for inclusion in the Monitoring program, including the approach to be applied to fire protection SSCs already included in the scope of the Maintenance Rule program, is described as follows:

Phase 1 – Scoping

The following categories of SSCs and programmatic elements will be reviewed during the implementation phase for inclusion in the NFPA 805 monitoring program:

- SSCs required to comply with NFPA 805, specifically:
  - Fire protection systems and features required by the NSCA.
  - Fire protection systems and features modeled in the Fire PRA.
  - Fire protection systems and features required by Chapter 3 of NFPA 805.
  - Nuclear Safety Capability Assessment equipment (for the purposes of NFPA 805 Monitoring, "NSCA equipment" includes NSCA equipment, Fire PRA equipment, and NPO equipment).
  - Structures, systems and components relied upon to meet radioactive release criteria.
- Fire Protection Program elements
- Key assumptions in engineering analyses (specifically analyses performed to demonstrate compliance with the nuclear safety and radioactive release performance criteria)

As a minimum, the fire protection systems and features (required to meet Chapter 3 of NFPA 805 and the NSCA criteria) and SSCs required to meet the radioactive release criteria will be included in the existing inspection and test program and system/program health program. In addition passive features (barriers, drains, curbs, etc.) that are relied upon to demonstrate compliance with Chapter 4 of NFPA 805 will also be included in the existing inspection and test program and system/program health program. Once applicable updates are completed, the existing programs will be adequate for routine monitoring of these SSCs.

Phase 2 – Screening Using Risk Criteria

The equipment from Phase 1 scoping will be screened to determine the appropriate level of NFPA 805 monitoring. As a minimum, the SSCs identified in Phase 1 will be part of an inspection and test program and/or system/program health reporting process. If not in the current program, the SSCs will be added in order to assure that the criteria can be met reliably.

1. Screening of Fire Protection Systems and Features:

Those fire protection systems and features identified in Phase 1 are candidates for additional monitoring in the NFPA 805 program commensurate with risk significance.

The Fire PRA is the primary tool used to establish the risk significance criteria and performance bounding guidelines. The screening thresholds used to determine risk significant analysis units will be those that meet the following criteria:

Risk Achievement Worth (RAW) of the monitored parameter ≥ 2.0

AND either

Core Damage Frequency (CDF) x (RAW) ≥ 1.0E-7 per year

OR

Large Early Release Frequency (LERF) x (RAW) ≥ 1.0E-8 per year

CDF, LERF, and RAW (monitored parameter) are calculated for each fire area. The "monitored parameter" will be established at a level commensurate with the amenability of the parameter to risk measurement (e.g., a fire barrier may be more conducive to risk measurement than an individual barrier penetration).

Fire protection systems and features that meet or exceed the criteria identified above are considered High Safety Significant (HSS) and will be evaluated for inclusion in the NFPA 805 performance monitoring program. The remaining required fire protection systems and features will be monitored in accordance with existing inspection and test programs and in the existing system/program health program and fire impairment processes and procedures.

2. Screening of Nuclear Safety Capability Assessment (NSCA) Equipment:

Required NSCA equipment identified in Phase 1 (excepting equipment within the scope of Non-Power Operations or NPO) will be screened for safety significance using the Fire PRA and the Maintenance Rule Scope and Basis guidelines which differentiate HSS equipment from Low Safety Significance (LSS) equipment. HSS NSCA equipment not currently monitored in Maintenance Rule will be evaluated for inclusion in the NFPA 805 performance monitoring program. All NSCA equipment not HSS will be considered LSS and will not be included in the monitoring program beyond normal inspection and test programs and system/program health reporting processes and procedures.

For NPO modes, quantitative measures of the effectiveness of fire prevention to manage fire risk during Higher Risk Evolutions are not feasible. Therefore, fire risk management effectiveness will be monitored programmatically similar to combustible material control and other fire prevention program processes. Additional monitoring beyond inspection and test programs or system/program health reporting will not be necessary to effectively assess fire risk management effectiveness during NPO modes.

3. Screening of SSCs Relied upon for Radioactive Release Criteria

Since the evaluations performed to meet the radioactive release performance criteria are qualitative, the SSCs relied upon to meet the radioactive release performance criteria are not amenable to quantitative risk measurement. Additionally, since 10CFR Part 20 limits (which are lower than releases due to core damage and containment breach) for radiological

effluents are not being exceeded, equipment relied upon to meet the radioactive performance criteria is considered inherently low risk. Therefore, additional monitoring beyond inspection and test programs and system/program health reporting is not considered necessary.

b. *Assigning availability, reliability, and performance goals to SSCs:*

The process that will be used to assign availability, reliability, and performance goals to High Safety Significant (HSS) SSCs within the scope of the monitoring program is known as Phase 3 – "Risk Target Value Determination".

Phase 3 establishes the target values for reliability and availability for the fire protection systems and features that met or exceeded the screening criteria and the HSS NSCA equipment identified in Phase 2.

Target values for reliability and availability for the fire protection systems and features are established at the component level, program level, or functionally through the use of the pseudo-system or "performance monitoring group" (PMG) concept. The actual action level is determined based on the number of component, program, or functional failures within a sufficient bounding time period (2 to 3 operating cycles).

HSS NSCA equipment–specific performance criteria will be evaluated for inclusion in the NFPA 805 Performance Monitoring program, provided the criteria are consistent with Fire PRA assumptions.

The action level threshold for reliability and availability will be no lower than the fire PRA assumptions. Adverse trends and unacceptable levels of availability, reliability, and performance will be reviewed against these action levels. The Monitoring Program failure criteria and action level targets will be documented in the NFPA 805 Monitoring Program Engineering Evaluation.

Fire protection systems and features, NSCA equipment, SSCs required to meet radioactive release criteria and fire protection program elements that do not meet the screening criteria in Phase 2 will be included in existing inspection and test programs and system/program health programs. Reliability and availability criteria will not be assigned.

Low Safety Significant (LSS) SSC's do not specifically require assignment of availability, reliability, and performance goals. Programmatic elements not readily quantified will be evaluated using the existing program health process embodied in Waterford procedures such as EN-DC-143.

c. To specifically address programmatic elements which fail to meet performance goals, a qualitative process using the existing health programs will be applied. Fire protection health reports, self-assessments, regulator and insurance (NEIL) reports provide inputs to this monitoring program, as does the Corrective Action Program (CAP) delineated in Waterford procedure EN-LI-102. Performance goal deficiencies thus identified will be entered and resolved in accordance with the CAP.

d. Fundamental Fire Protection program elements, such as combustible materials, ignition sources, impairments and compensatory measures, and fire brigade performance, are qualitative in nature and not amenable to numerical methods to derive reliability and availability. These program elements will be monitored in accordance with existing inspection and test programs and in the existing system/program health program and fire impairment processes and procedures.

e. Waterford 3 is not planning to make use of EPRI Technical Report 1006756 at this time and will delete references to the report in the LAR.

f. Periodic assessments of the Monitoring program will be performed approximately every 2 to 3 operating cycles taking into account, where practical, industry operating experience. This periodic assessment is included as a part of the Monitoring program implementation (LAR implementation item S2-10). The assessments will be conducted as a part of other established assessment activities, and will include these elements:

- Review systems with performance criteria. Do performance criteria still effectively monitor the functions of the system?
- Do the criteria still monitor the effectiveness of the fire protection and nuclear safety capability assessment systems?
- Have the supporting analyses been revised such that the performance criteria are no longer applicable or new fire protection and nuclear safety capability assessment SSCs, programmatic elements and/or functions need to be in scope?
- Based on the assessment period, are there any trends in monitored elements that should be addressed that are not being addressed?

g. NFPA 805 audits and assessments will be conducted under the Waterford 3 Quality Assurance and Fire Protection Programs and procedures. Currently, Section C.2 of the Waterford QAPM specifies that a fire protection and loss prevention program inspection and audit shall be performed using either off-site licensee personnel or an outside fire protection firm at least once every 24 months.

PROGRAMMATIC (PROG)

RAI PROG 01

NFPA 805 Section 2.7.1.1 requires that "the analyses performed to demonstrate compliance with this standard shall be documented for each nuclear power plant (NPP). The intent of the documentation is that the assumptions be clearly defined and that the results be easily understood, that results be clearly and consistently described, and that sufficient detail be provided to allow future review of the entire analyses. Documentation shall be maintained for the life of the plant and be organized carefully so that it can be checked for adequacy and accuracy either by an independent reviewer or by the AHJ."

NFPA 805 Section 2.4.3.3 requires that "the PSA approach, methods, and data shall be acceptable to the AHJ. They shall be appropriate for the nature and scope of the change being evaluated, be based on the as-built and as-operated and maintained plant, and reflect the operating experience at the plant."

NFPA 805 Section 3.3.1.2 requires that "procedures for the control of general housekeeping practices and the control of transient combustibles shall be developed and implemented."

Fire PRA analyses assume combustible loading will be maintained at or below certain values. Please provide a description of how the combustible controls program will be administered to ensure that FPRA assumptions regarding combustible loading will be met.

Waterford 3 Response

EN-DC-161 (Control of Combustibles) is currently utilized for controlling any new combustibles in any fire area. This procedure will be updated as necessary to ensure that fire PRA assumptions regarding combustible loading will be met. Where pinch points exist, they will receive special treatment where additional controls will be utilized (as described in RAI SSA 10) including possibly changes to locations into Level 1 areas per EN-DC-161. A Level 1 area requires a Transient Combustible Evaluation (TCE) from Fire Protection be processed prior to introduction of new combustible materials.

RAI PROG 02

NFPA 805. Section 3.2.3, "Procedures" states that "Procedures shall be established for implementation of the fire protection program. In addition to procedures that could be required by other sections of the standard, the procedures to accomplish the following shall be

established: (1) Inspection, testing, and maintenance for fire protection systems and features credited by the fire protection program ..."

Please provide a description of how the guidance in EPRI Technical Report 1006756, "Fire Protection Equipment Surveillance Optimization and Maintenance Guide," or other performance-based approach will be used to adjust the frequency of inspection, test, and maintenance activities.

Such changes, permitted by NFPA 805 Section 3.2.3 require NRC approval with an appropriate justification and request in accordance with 10 CFR 50.48(c)(2)(vii).

Waterford 3 Response

Waterford 3 will not implement, at this time, performance based methods for establishing frequencies for inspection, testing, or maintenance for fire protection systems and features

credited by the fire protection program. Should Waterford decide in the future to implement such a performance based approach, required NRC approval will be obtained or an NRC approved methodology will be utilized. Therefore, the “Compliance Statement” for Section 3.2.3 “Procedures” will change from “Complies with Clarification” to “Complies” and the reference to “EPRI TR-1006756, Fire Protection Equipment Surveillance Optimization and Maintenance Guide” will be deleted.

RAI PROG 03

NFPA 805 Section 2.7.3, "Quality" states that:

> *(2.7.3.1) "Review. Each analysis, calculation or evaluation performed shall be independently reviewed."*
>
> *(2.7.3.2) "Verification and Validation. Each calculational model or numerical method used shall be verified and validated through comparison to test results or comparison to other acceptable models."*
>
> *(2.7.3.3) "Limitations of Use. Acceptable engineering methods and numerical models shall only be used for applications to the extent these methods have been subject to verification and validation. These engineering methods shall only be applied within the scope, limitations, and assumptions prescribed for that method."*
>
> *(2.7.3.4) "Qualification of Users. Cognizant personnel who use and apply engineering analysis and numerical models (e.g., fire modeling techniques) shall be competent in that field and experienced in the application of these methods as they relate to nuclear power plants, nuclear power plant fire protection, and power plant operations."*
>
> *(2.7.3.5) "Uncertainty Analysis. An uncertainty analysis shall be performed to provide*

The licensee stated that:

a. *Analyses, calculations, and evaluations performed in support of compliance with 10 CFR 50.48(c) were performed in accordance with Entergy procedures that require independent review.*
b. *Calculational models and numerical methods used in support of compliance with 10 CFR 50.48(c) were verified and validated as required by Section 2.7.3.2 of NFPA 805.*
c. *Engineering methods and numerical models used in support of compliance with 10 CFR 50.48(c) were applied appropriately as required by Section 2.7.3.3 of NFPA 805.*
d. *Cognizant personnel who use and apply engineering analysis and numerical methods in support of compliance with 10 CFR 50.48(c), are competent and experienced as required by Section 2.7.3.4 of NFPA 805.*
e. *During the transition to 10 CFR 50.48(c), work was performed in accordance with the quality requirements of Section 2.7.3 of NFPA 805.*
f. *That uncertainty analyses were performed as required by 2.7.3.5 of NFPA 805 and the results were considered in the context of the application. This is of particular interest in fire modeling and Fire PRA development.*

The LAR states that while analyses supporting the Transition Report have been performed in accordance with the quality requirements of Section 2.7.3 of NFPA 805, no specific commitment has been made to comply with these requirements for future analyses. Please provide this commitment or define any alternative requirements that will be used for future analyses.

Additionally, Section 4.5.1.2, "FPRA Quality," of the Transition Report states that fire modeling was performed as part of the Fire PRA development (NFPA 805 Section 4.2.4.2). This requires that qualified fire modeling and PRA personnel work together.

Furthermore, Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," of the Transition Report states: that cognizant personnel who use and apply engineering analysis and numerical methods in support of compliance with 10 CFR 50.48(c), are competent and experienced as required by Section 2.7.3.4 of NFPA 805.

During the transition to 10 CFR 50.48(c), work was performed in accordance with the quality requirements of Section 2.7.3 of NFPA 805. Personnel who used and applied engineering analysis and numerical methods (e.g. fire modeling) in support of compliance with 10 CFR 50.48(c) are competent and experienced as required by NFPA 805 Section 2.7.3.4.

Post-transition, for personnel performing fire modeling or Fire PRA development and evaluation, Waterford 3 will develop and maintain qualification requirements for individuals assigned various tasks. Position Specific Guides will be developed to identify and document required training and mentoring to ensure individuals are appropriately qualified per the requirements of NFPA 805 Section 2.7.3.4 to perform assigned work. See Attachment S for an Implementation Item."

Regarding qualifications of users of engineering analyses and numerical models:

a. *Please describe what constitutes the appropriate qualifications for the Waterford 3 staff and consulting engineers to use and apply the methods and fire modeling tools included in the engineering analyses and numerical models.*
b. *Please describe the process/procedures for ensuring the adequacy of the appropriate qualifications of the engineers/personnel performing the fire analyses and modeling activities.*
c. *Please explain the communication process between the fire modeling analysts and PRA personnel to exchange the necessary information and any measures taken to assure the fire modeling was done adequately.*

<u>Waterford 3 Response</u>

Regarding qualifications of users of engineering analyses and numerical models(a, b, & c):

From the RAI, “The LAR states that while analyses supporting the Transition Report have been performed in accordance with the quality requirements of Section 2.7.3 of NFPA 805, no specific commitment has been made to comply with these requirements for future analyses. Provide this commitment or define any alternative requirements that will be used for future analyses. “

LAR Attachment S has two implementation items documenting commitments to adhere to the requirements of NFPA 805 section 2.7.3 (implementation items S2-14 and S2-15). While the specific details of qualification, processes, and procedures are not yet fully developed, the commitments in Attachment S ensure Waterford 3 will meet the requirements of NFPA 805 section 2.7.3.

Current Entergy PRA staff members are required to maintain qualification (qual) cards (these are common in the industry). These qualification cards are maintained to ensure PRA personnel have the appropriate training and technical expertise to perform assigned work. The contents of the these qualification cards, along with project specific training are the type of items that will be fully developed and explained in the process of meeting the documented implementation items in LAR Attachment S (items S2-14 and S2-15).

The following items list specific aspects of NFPA 2.7.3 and show the controls Waterford 3 currently has in place to ensure quality and compliance.

NFPA 805 Section 2.7.3.1 – Review-
Per EN-DC-134 (Design Verification), design verification should be performed on all quality related and augmented quality related design activities and documents as required by the Entergy QAPM and for Engineering Changes as required by EN-DC-115.

NFPA 805 Section 2.7.3.2 – Verification and Validation
Per EN-DC-126 (Engineering Calculation Process), design verification is performed on quality related calculations in accordance with requirements in procedure EN-DC-134, Design Verification. Augmented Quality related calculations (FREs are an example) are also covered by EN-DC-126.

NFPA 805 Section 2.7.3.3 – Limitations of Use-
Per EN-DC-134 – The purpose of this procedure provides the methods and requirements for performing design verification of quality related and augmented quality related documents.

NFPA 805 Section 2.7.3.4 – Qualification of Users-
Waterford 3 will develop and maintain qualification requirements for individuals assigned various tasks. Position Specific Guides will be developed to identify and document required training and mentoring to ensure individuals are appropriately qualified per the requirements of NFPA 805 Section 2.7.3.4 to perform assigned work. See Attachment S for an Implementation Item.

NFPA 805 Section 2.7.3.5 – Uncertainty Analysis-
Uncertainty evaluations for the various tasks used to develop the FPRA model (specifically those outlined in NUREG/CR-6850) were completed. A summary of the uncertainty is provided in PRA-W3-05-007 (FPRA Summary Report). While many of the uncertainty analyses were qualitative, some quantitative uncertainty results are presented. FPRA work done as part of the transition or needed to maintain the program (i.e. FPRA model update) will include uncertainty evaluations. The exact type and scope of the uncertainties for future work is not known, but Waterford will follow the necessary guidance (examples included - NUREG/CR-6850, FAQs, PRA Standard) in the development of FPRA related uncertainty analysis.

FIRE PROTECTION ENGINEERING (FPE)

RAI FPE 01

LAR, Section 6.0, "References," is missing reference codes and/or code editions that are in LAR. Table 8-1. It is not clear if the LAR contains the complete list of reference codes required. Confirm whether or not the LAR contains a complete list of required codes. Please clarify why there is a difference between the LAR Section 6.0 and the LAR, Table 8-1 and update the necessary LAR sections as appropriate to reflect a complete list. For example:

a. *Table 8-1,3.8.1 refers to NFPA 72 (1975) but LAR 6.0 does not list NFPA 72*
b. *Table 8-1,3.8.1(2) refers to NFPA 720 (1975) but LAR 6.0 does not list NFPA 720*
c. *Table 8-1, 3.8.1 (1) refers to NFPA 72E (1974) but LAR 6.0 (6.13) refers to 1975 edition*
d. *Table 8-1, 3.3.1.3.1 refers to NFPA 518 but neither Table 8-1 nor LAR 6.0 (6.11) contains the edition*
e. *Table 8-1, 3.3.1.3.1 refers to NFPA 241 but neither Table 8-1 nor LAR 6.0 (6.17) contains the edition*
f. *Table 8-1, 3.3.3 refers to NFPA 101 but neither Table 8-1 (Reference Document column) nor LAR 6.0 list NFPA 101 nor contain the edition*
g. *Table 8-1, 3.3.2 refers to NFPA 220 but neither Table 8-1 (Reference Document column) nor LAR 6.0 list NFPA 220 nor contain the edition*
h. *Table 8-1,3.3.5.1 refers to IEEE-383 or NFPA 262 but neither Table 8-1 nor LAR 6.0 contains either code edition; LAR 6.0 does not list NFPA 262 or IEEE 383 as applicable*

Please clarify the remaining Table 8-1 references and their editions and ensure they are captured with Section 6.0 and Table 8-1 appropriately and provide justification for not including any 8-1 referenced codes in LAR, Section 6.0.

Waterford 3 Response

A Waterford 3 review of references listed in LAR, Section 6.0 associated with Table B-1 confirmed the issues identified above. The extent to which Waterford 3 complies with codes listed in Section 6.0 is described in the applicable section of Table B-1 or other sections/attachments of the LAR. Provided below is a listing of LAR, B-1 Table code reference issues and their clarification.

RAI #	LAR Attachment A (B-1 Table)	LAR 6.0 (References)
01a	Section 3.8.1: The correct code is NFPA 72D, 1975 Edition.	Add NFPA 72D, 1975 Edition.
01b	Section 3.8.1(2): NFPA 72D, 1975 Edition is correct.	Add NFPA 72D, 1975 Edition.
01c	Section 3.8.1(1): NFPA 72E, 1974 Edition is correct.	LAR 6.13: change to reflect the edition is 1974.

RAI #	LAR Attachment A (B-1 Table)	LAR 6.0 (References)
01d	Section 3.3.1.3.1: The code edition for NFPA 51B is 1999.	LAR 6.11: change to reflect the edition is 1999.
01e	Section 3.3.1.3.1: The code edition for NFPA 241 is the 2000 Edition. NFPA 241, Section 5.1 is the only portion of this code that addresses "Hot Work". Also a new LAR Attachment A "Confirmatory Item" (VFDR 3.3.1.3.1) and Attachment S item will be added to develop or revise plant procedures/documents to address requirements of NFPA 241, Section 5.1 for Thermit Welding (Cad Welding).	LAR 6.17: change to reflect the edition is 2000 as well as specifying that only Section 5.1 is applicable.
01f	Section 3.3.3: Add NFPA 101, 2000 Edition. Note: NFPA 101, Sections 10.2.3 and 10.2.7 are the only portions of this code that are applicable to the requirements specified in Table B-1, Section 3.3.3.	Add NFPA 101, 2000 Edition, Section 10.2.3 and 10.2.7.
01g	Section 3.3.2: Add NFPA 220, 1999 Edition. Note: NFPA 220, Section 3.3.4 is the only portion of this code that is applicable to the requirement specified in Table B-1, Section 3.3.2.	Add NFPA 220, 1999 Edition, Section 3.3.4.
01h	Section 3.3.5.1: The editions are 1974 for IEEE-383 and 2007 for NFPA 262.	Add IEEE-383, 1974 Edition and NFPA 262, 1999 Edition.

Additional clarifications for Table B-1 and LAR Section 6.0 are as follows:

LAR Attachment A (B-1 Table)	LAR 6.0 (References)
Section 3.3.1.2(6): This section references NFPA 55 but does not contain the edition. The code edition for NFPA 55 is 1998.	Add NFPA 55, 1998 Edition.
Section 3.8.2 refers to NFPA 72D and NFPA 72 E but does not contain the edition for either. The code edition for NFPA 72D is 1975 and 1974 for NFPA 72 E.	Add NFPA 72D, 1975 Edition and NFPA 72E, 1974 Edition.
Section 3.3.5.3 refers to NFPA 262 but Table B-1 (Reference Document column) does not list NFPA 262 nor contain the edition. Add NFPA 262, 2007 Edition.	Add NFPA 262, 1999 Edition.
Section 3.3.6 refers to NFPA 256 but Table B-1 (Reference Document column) does not list NFPA 256 nor contain the edition. Add NFPA 256, 1998 Edition.	Add NFPA 256, 1998 Edition.
Section 3.9.1: Correctly identifies the 1973 edition of NFPA 15 as the code of record for Waterford 3.	LAR 6.5 conflicts with Section 3.9.1 and lists the code of record as the 1976 Edition. Change the edition of NFPA 15 to1973.

The above clarifications ensure references and their editions are captured with Section 6.0 and Table B-1 appropriately.

RAI FPE 02

Table 8-1 :3.3.9: There is a reference to Attachment S regarding VFDR #3.3.9-1 " ... to require periodic inspection of transformer oil collection basins and drain paths to ensure that they are free of debris and capable of performing their design function." Clarify the following:

a. *Please describe the frequency used and the justification (e.g. documents used or referenced) for the basis of this frequency of inspections and what the inspection tasks will be.*
b. *Please describe how this frequency is related to the fire protection system inspection frequencies.*
c. *Please describe how the inspections will be monitored and how the inspection frequency changes will be monitored.*

Waterford 3 Response

a. Describe the frequency used and the justification (e.g. documents used or referenced) for the basis of this frequency of inspections and what the inspection tasks will be:

   Transformer rock filled oil collection basins will be inspected in accordance with the guidance in IEEE Standard 980-1994 “IEEE Guide for Containment and Control of Oil Spills in Substations”. The inspection frequency will be based on the transformer outage during each refueling outage and will ensure there is no filling of void spaces by dirt, dust, or silt.

b. Describe how this frequency is related to the fire protection system inspection frequencies:

   The Transformer rock filled oil collection basin inspection will be performed concurrent with the transformer suppression system inspections performed during each transformer/refueling outage.

c. Describe how the inspections will be monitored and how the inspection frequency changes will be monitored:

   The oil collection basin inspection procedure will require inspection results be transmitted to Fire Protection Engineering for performance monitoring purposes and procedure changes will require Fire Protection review/concurrence. A change to the frequency of this procedure would be a fire protection program change and is controlled by Procedure EN-DC-128 “Fire Protection Impact Reviews”. This procedure ensures that qualified fire protection engineering personnel determine if the change requires prior NRC approval and if not, determine the acceptability of the change based on compliance with the approved Waterford 3 Fire Protection Program.

RAI FPE 03

Table 8-1: 3.4.2 and 3.4.3 WF3 states "complies" without any required action.

Please clarify that any Radiation Release components (from Attachment E) have been incorporated into the fire pre-plans, drills, and training. There is a reference to FP-001-020, Rev. 304 as the procedure "providing adequate coordination with other plant groups .... " The

procedure did not clearly describe the needed NFPA 805 radiation release actions and instructions/guidance. Please clarify if FP-001-020 includes the required information to meet the NFPA 805 radiation release performance criteria and clarify if the group which will handle these criteria is clearly identified in the procedure. If this procedure is to be updated, please identify the respective Implementation Item.

Waterford 3 Response

Section 4.0 in the Pre-Fire Strategies on "Radiological Hazards" contains radiological information, which includes radiological material, the possibility of surface contamination and dose rates in the area. The engineering change package (EC15965), which has been issued includes the updates to the Pre-Fire Strategies (PFS) for Radiological Release for NFPA-805. All of EC-15965 post actions have been updated including FP-001-020 and pre-fire strategies.

Procedure FP-001-020 Fire Emergency/Fire Report" sections 4.5, 6.4.2.4 and 6.9 provide for Radiation Protection (RP) to support the Fire Brigade. This is an administrative procedure. Special instructions for radiological procedures and precautions are found in Fire Brigade Training and in the Pre-Fire Strategies.

Procedure FP-001-020 clearly identifies RP as the group responsible for radiological control and support of the Fire Brigade. Specialized procedures for radiological sampling and spill control are contained within the RP department (See attachment E of the LAR for the initial procedures performed for radiological support of air and water sampling).

RAI FPE 04

During the audit, it was noted that an addressable Pyrotronics fire alarm system was installed around 1995; however, the LAR references the 1972 editions of NFPA 720 and E as the code of record. Per NFPA 805, Section 1.8, Code of Record, Please provide further justification for why the current edition of NFPA 72 (as of time of design submission) was not referenced and utilized when this system was installed.

Waterford 3 Response

The Waterford 3 code of record for NFPA 72D and NFPA 72E are 1975 and 1974 respectively (incorrectly reported above as 1972 editions). Refer to Waterford 3 response to Fire Protection Engineering RAI 01(a) and (b) for clarifications.

The original fire alarm system at Waterford 3 was installed during construction in accordance with NFPA 72D, 1975 edition and 72E, 1974 edition (refer to UFSAR Sections 9.5.1.1.4.17 and 9.5.1.1.4.18) with one exception. The exception was equipment used in the fire alarm system be tested and accepted by a nationally recognized testing laboratory. Waterford 3 SER, Supplement 5 (dated June 1983), Section 9.5.1.2(3) documented the staff review of the design and the fire detection and control system, along with the manufacturer's own testing standards and test results. The Staff concluded that the panels would perform in an acceptable manner, consistent with the requirements of NFPA Standards 72D and 72E. This NRC acceptance stipulated that Waterford 3 commit to obtain Factory Mutual (FM) approval and implement any equipment modifications required to obtain FM approvals. The manufacture of the original fire alarm system equipment did not obtain FM approval. As a result, the entire fire alarm system was replaced by a state of the art automatic fire detection system in the mid 1990's. Equipment used in the new fire alarm system is Underwriters Laboratories (UL) listed and/or FM approved

for its intended service which conforms to the commitment specified in Waterford 3 SER, Supplement 5, Section 9.5.1.2(3). Waterford 3 installed the new fire alarm system in accordance the fire protection program licensing basis approved which was NFPA 72D, 1975 edition and 72E, 1974 edition.

Waterford 3 evaluations documenting the code of record for NFPA 72D and 72E for conformance with NFPA 805, Section 1.8 are provided in Attachments 7.1 in Engineering Reports No. WF3-FP-010-00006 and WF3-FP-010-00007 respectively. NFPA 805, Section 1.8 specifies that the code of record be either 1) The edition of the code or standard in effect at the time the fire protection systems or feature was designed; 2) Specifically committed to the authority having jurisdiction. In summary, both evaluations concluded that Waterford 3 committed to the 1975 Edition of NFPA 72D as identified in UFSAR Section 9.5.1.1.4.17 and the 1974 Edition of NFPA 72E as identified in UFSAR Section 9.5.1.1.4.18. Therefore, in compliance with NFPA 805 – 2001 Edition, Section 1.8 the "Code of Record" for NFPA 72D and 72E is 1975 and 1974 respectively.

RAI FPE 05

There are several areas with HEMYC fire wrap installed. Please clarify whether any HEMYC is credited to meet NFPA 805. If so, describe which areas of the HEMYC is credited and why. There is also some 3M fire wrap installed. Clarify whether all of the 3M fire wrap is credited for NFPA 805. Also, Please provide further description for any other electrical raceway fire barrier system (ERFBS) fire wraps (e.g., pyrocrete, thermolag, etc.) currently installed (type and fire area location) and whether or not it is credited for NFPA 805.

Waterford 3 Response

Electrical raceway fire barrier system necessary for compliance with NFPA 805 and the areas where they are required are described in LAR, Attachment A (Table B-1), Section 3.11.5.

The HEMYC fire wrap system is not utilized as an ERFBS to meet NFPA 805 separation requirements. HEMYC fire wrap has been and will continue to be utilized to meet electrical separation requirements in Regulatory Guide 1.75.

Waterford 3 currently utilizes 3M Interam™ ERFBS in Fire Areas RAB-17 (1-hour rated) and RAB 30 (3-hour rated) to meet 10CFR50, Appendix R, Section III.G.2 separation requirements. The 3M Interam™ ERFBS in Fire Area 30 is not credited for NFPA 805. The 3M Interam™ ERFBS in Fire Area 17 is credited for NFPA 805.

HEMYC and 3M Interam™ fire wrap systems are the only ERFBS's installed at Waterford 3. As stated above none of the HEMYC ERFBS is credited for NFPA 805. LAR, Attachment A, Section 3.11.5 identifies that ERFBS's credited for NFPA 805 are only necessary in Fire Areas RAB-2, 5, 6, and 17. All ERFBS in these fire areas credited for NFPA 805 utilize or will utilize the 1-hour rated 3-M Interam™ ERFBS.

RAI FPE 06

For each of the eight approval requests listed in Attachment L, Please provide further justification and clarification for each safety margin and DID discussion. For example, the

current DID discussions identify what the DID requirements are with only a simple statement that DID has been satisfied. For each approval request, please provide a discussion and further justification for how each DID echelon is met and how each safety margin is satisfied

Waterford 3 Response

Attachment L, Approval Request 1 is reworded as follows:

Approval Request 1

Safety Margin and Defense-in-Depth

Compensatory actions specified in Technical Requirements Manual (TRM) 3.7.10.2 are implemented in areas with suppression when the suppression system is impaired/disabled due to hot work operations in the area with suppression. These compensatory actions provide an equivalent level of protection as the suppression system being impaired/disabled. Fire watch personnel are assigned continuous duty with the personnel performing Hot Work and additional precautions are taken to ensure combustibles are not ignited by hot work activities in accordance with Entergy procedure EN-DC-127 “Control of Hot Work and Ignition Sources”. Additional compensatory measures, as described above, are taken when hot work activities are necessary in areas where the suppression system is isolated. These compensatory measures provide an equivalent level of protection as would be provided by the suppression system. Therefore, the safety margin is maintained.

The three echelons of defense-in-depth are 1) prevent fires from starting (administrative procedures for combustible/hot work controls), 2) rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic/manual fire suppression, fire brigade/pre-fire plans), and 3) provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, success path remains free of fire damage, recovery actions).

Impaired/disabled sprinkler systems do not affect echelon 1 of the defense-in-depth concept because sprinkler systems are not involved with administrative procedures to prevent fire from occurring. echelons 2 and 3 of the defense-in-depth concept are maintained by implementation of TRM required compensatory actions that provide an equivalent level of protection, as the suppression system, when hot work is required in an area where the suppression system is impaired/disabled. Therefore, echelons 2 and 3 of the defense in depth concept are maintained.

Attachment L, Approval Request 2 is reworded as follows:

Approval Request 2

Safety Margin and Defense-in-Depth

Exposed non-plenum rated electrical wiring located above suspended ceilings is limited, sufficiently dispersed, considered an insignificant fire hazard, installed in accordance with Regulatory Guide 1.75, and is not capable of causing fire damage to components necessary for safe shutdown. Therefore, safety margin inherent in the analysis for the fire event has been preserved.

The three echelons of defense-in-depth are 1) prevent fires from starting (administrative procedures for combustible/hot work controls), 2) rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic/manual fire suppression, fire brigade/pre-fire plans), and 3) provide adequate level of fire protection for systems and

structures so that a fire will not prevent essential safety functions from being performed (fire barriers, success path remains free of fire damage, recovery actions).

Exposed non-plenum rated electrical wiring located above suspended ceilings do not affect echelon 1 of the defense-in-depth concept because it is not involved with administrative procedures to prevent fire from occurring. The limited quantity of this wiring above suspended ceilings is sufficiently dispersed, is considered an insignificant fire hazard, and is not capable of causing fire damage to components necessary for safe shutdown. Electrical wiring at Waterford 3 is installed in accordance with Regulatory Guide 1.75; therefore, a fire originating in this non-plenum rated, non safety related wiring will not adversely affect wiring required for safety related systems. Therefore, echelons 2 and 3 of the defense in depth concept are maintained.

Attachment L, Approval Request 3 is reworded as follows:

Approval Request 3

Safety Margin and Defense-in-Depth

Use of PVC conduits in embedded concrete or buried underground applications is not capable of causing fire damage to components necessary for safe shutdown because electrical circuits within these raceways are not subject to damage from exposure fire and fire internal to the conduit will not expose or propagate to expose external targets. Therefore, the safety margin inherent in the analysis for the fire event has been preserved.

The three echelons of defense-in-depth are 1) prevent fires from starting (administrative procedures for combustible/hot work controls), 2) rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic/manual fire suppression, fire brigade/pre-fire plans), and 3) provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, success path remains free of fire damage, recovery actions).

Use of PVC conduits in embedded concrete or buried underground applications do not affect echelons 1 or 2 of the defense-in-depth concept because the use of PVC conduits is not involved with administrative procedures to prevent fire from occurring or detection, control, and extinguishment of fire. Electrical circuits within these raceways are not subject to damage from exposure fire and fire internal to the conduit will not expose or propagate to expose external targets. Therefore, an adequate level of fire protection for systems and structures is provided so that a fire will not prevent essential safety functions from being performed. Therefore, echelon 3 of the defense-in-depth concept is maintained.

Attachment L, Approval Request 4 is reworded as follows:

Approval Request 4

Basis for Request

The basis for the approval request of this deviation is:

- Video/communication/data cables are low voltage and do not pose a fire hazard because they are not susceptible to self ignition and electrical shorts that could result in a fire.
- With the exception of the communications room located on elevation +7 in Fire Area RAB-27; the remaining areas contain a limited quantity of this wiring which is sufficiently dispersed, is considered an insignificant fire hazard, and is not capable of causing fire damage to components necessary for safe shutdown. Combustibles associated with these

cables are considered insignificant with regards to combustible loading in the affected areas due to the limited quantity of these cable types.

- The communications room, which contains a quantity of exposed cabling, represents a moderate fire loading (less than 2-hour fire duration) when compared to the floor area of the room. The room has automatic fire detection, manual fire hose and fire extinguisher coverage and is separated from other fire areas by 3-hour rated barriers and from other rooms within Fire Area RAB-27 by substantial concrete masonry walls and metal door. A fire in the communications room would be detected rapidly by the fire detection system and the present fire hazard is well within the capability of the plant fire brigade to control and extinguish.

- Electrical wiring at Waterford 3 is installed in accordance with Regulatory Guide 1.75; therefore a fire originating in this non-rated, non safety related wiring will not adversely affect wiring required for safety related systems.

- The substantial concrete masonry walls and metal door separating the communications room from other rooms within Fire Area RAB-27 is considered adequate to prevent fire propagation outside the room. However, if fire should propagate across this barrier, there are insufficient combustibles in surrounding areas to propagate the fire to damage redundant equipment required for safe shutdown.

Acceptance Criteria Evaluation

Nuclear Safety and Radiological Release Performance Criteria

Exposed video/communication/data cables, installed at Waterford 3, with cable construction that does not comply with a flame propagation test acceptable to the AHJ do not present a fire hazard capable of damaging components required for safe shutdown. Therefore, there is no impact on the nuclear safety performance criteria.

The radiological release review was performed based on the fire suppression activities in areas containing or potentially containing radioactive materials. The limited use of video/communication/data cabling has been shown acceptable and does not create or pose an un-acceptable fire hazard. Therefore, the radiological release performance criteria are satisfied based on the determination of limiting radioactive release.

Safety Margin and Defense-in-Depth

Exposed video/communication/data cables, installed at Waterford 3, with cable construction that does not comply with a flame propagation test acceptable to the AHJ is not capable of causing fire damage to components necessary for safe shutdown due to the insignificant fire hazard in areas other than the communications room located on elevation +7 in Fire Area RAB-27. The substantial concrete masonry walls and metal door separating the communications room from other rooms within Fire Area RAB-27 is considered adequate to prevent fire propagation outside the room. However, if fire should propagate across this barrier, there are insufficient combustibles in surrounding areas to propagate the fire to damage redundant equipment required for safe shutdown. Therefore the safety margin inherent in the analysis for fire event has been preserved.

The three echelons of defense-in-depth are 1) prevent fires from starting (administrative procedures for combustible/hot work controls), 2) rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic/manual fire suppression, fire brigade/pre-fire plans), and 3) provide adequate level of fire protection for systems and

structures so that a fire will not prevent essential safety functions from being performed (fire barriers, success path remains free of fire damage, recovery actions).

Exposed video/communication/data cables, installed at Waterford 3, with cable construction that does not comply with a flame propagation test acceptable to the AHJ does not affect echelon 1 of the defense-in-depth concept because cable construction is not involved with administrative procedures to prevent fire from occurring. In areas containing these cables which can not be categorized as insignificant; adequate detection, manual hose stream, and fire extinguishers are provided to ensure the fire is rapidly detected and controlled/extinguished by the fire brigade. Therefore, echelon 2 of the defense-in-depth concept is maintained. Fire Area RAB-27 (which includes the communications room) is separated from other fire areas by 3-hour rated barriers. The communications room is separated from other rooms within Fire Area RAB-27 by substantial concrete masonry walls and metal door. This separation is considered adequate to prevent fire propagation outside the room; however, if fire should propagate across this barrier, there are insufficient combustibles in surrounding areas to propagate the fire to damage redundant equipment required for safe shutdown. Therefore, echelon 3 of the defense-in-depth concept is maintained.

Based on the above discussion, the defense in depth philosophy is not adversely impacted.

Attachment L, Approval Request 5 is reworded as follows:

Approval Request 5

Safety Margin and Defense-in-Depth

The Reactor Coolant Pump (RCP) Oil Collection System, along with work control procedures controlling the addition of lube oil to the RCPs, will perform the design function as specified in NFPA 805, Section 3.3.12 with the deviations regarding reduced capacity of the oil collection tanks and remote oil fill lines not protected by the RCP Oil Collection System. The only safe shutdown equipment in the vicinity of the reactor coolant pumps is steam generator pressure indication, if this indication is lost due to fire, safe shutdown cooling can be monitored by pressure transmitters located on the main steam lines outside of containment. Therefore, the safety margin inherent in the analysis for the fire event will be preserved.

The three echelons of defense-in-depth are 1) prevent fires from starting (administrative procedures for combustible/hot work controls), 2) rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic/manual fire suppression, fire brigade/pre-fire plans), and 3) provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, success path remains free of fire damage, recovery actions).

Work control procedures controlling the addition of lube oil to the RCP's are consistent with fire prevention methods to preclude fires from starting during lube oil addition activities during power operations. These additional controls along with the administrative procedures ensure the echelon 1 defense-in-depth concept is maintained. The design of the RCP Oil Collection System is sufficient to ensure very limited quantities of escaping oil are postulated. A fire postulated for that condition would be characterized by the rapid consumption of the oil if ignited by hot surfaces and would not pose a significant fire hazard. The only safe shutdown component located in the vicinity of the RCPs is steam generator pressure indication. Should this pressure indication be lost due to fire conditions, there are pressure transmitters located outside of containment on the main steam lines that can be utilized for this indication. Therefore, safe shutdown would not be adversely affected and echelons 2 and 3 of the defense-in-depth concept are maintained.

Attachment L, Approval Request 6 is reworded as follows:

Approval Request 6

Safety Margin and Defense-in-Depth

Procedure FP-001-015 “Fire Protection System Impairments” identifies specific sprinkler systems (FP-M3A, FP-M4B, and/or FP-M22) that are impaired/inoperable based on which portion of the 4 inch fire main loop is out of service. Compensatory actions specified in Technical Requirements Manual (TRM) 3.7.10.2 are implemented for sprinkler system/systems determined to be inoperable. These compensatory actions provide an equivalent level of protection as the suppression system being impaired/disabled. Therefore, the safety margin inherent in the analysis for the fire event has been preserved.

The three echelons of defense-in-depth are 1) prevent fires from starting (administrative procedures for combustible/hot work controls), 2) rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic/manual fire suppression, fire brigade/pre-fire plans), and 3) provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, success path remains free of fire damage, recovery actions).

Impaired/disabled sprinkler systems do not affect echelon 1 of the defense-in-depth concept because sprinkler systems are not involved with administrative procedures to prevent fire from occurring. Sprinkler Systems FP-M3A, FP-M4B, and/or FP-M22 are installed to control and extinguish fires that do occur, thereby limiting damage and to provide an adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed. Compensatory actions that provide an equivalent level of protection as the suppression system being impaired/disabled (TRM 3.7.10.2) are implemented in accordance with Procedure FP-001-015 “Fire Protection System Impairments” for these sprinkler systems when the fire water supply demand can not be met due to the least demanding portion of the 4 inch fire main loop being out of service. Therefore, echelons 2 and 3 of the defense-in-depth concept are maintained.

Attachment L, Approval Request 7 is reworded as follows:

Approval Request 7

Safety Margin and Defense-in-Depth

Fire pump check valves and suction/discharge indicating gate valves are manufactured to ANSI Class 150 standards and have considerable margin with regards to pressure and temperature in the application they are being used. The valves are adequately designed for the fire water system at Waterford and have no adverse effect on the ability of the fire water system to perform its intended function. Therefore the safety margin inherent in the analysis for fire event has been preserved.

The three echelons of defense-in-depth are 1) prevent fires from starting (administrative procedures for combustible/hot work controls), 2) rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic/manual fire suppression, fire brigade/pre-fire plans), and 3) provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, success path remains free of fire damage, recovery actions).

Fire pump check valves and gate valves are not part of echelon 1 of the defense-in-depth concept because they are not involved with administrative procedures to prevent fire from occurring. They are part of echelons 2 and 3 of the defense-in-depth concept. The non-

approved/listed valves are manufactured to meet design requirements for pressure, materials, and temperature in excess of those required for the fire water system at Waterford. In addition, review of the plant corrective action program and surveillance test performance for these components reveals the equipment has performed satisfactorily. Therefore, the valves are adequately designed and have no adverse affect on the ability of the fire water system to perform its intended function of providing an adequate fire water supply. Therefore, echelons 2 and 3 of the defense in depth concept are maintained.

Attachment L, Approval Request 8 is reworded as follows:

Approval Request 8

Safety Margin and Defense-in-Depth

The use of the fire protection water for this non-fire protection system water demand has no adverse impact on the ability of the fire water system to provide required flow and pressure for the largest regulatory required sprinkler system, including fire hose demand, for a minimum of 2-hours. Therefore, the safety margin inherent in the analysis for fire event has been preserved.

The three echelons of defense-in-depth are 1) prevent fires from starting (administrative procedures for combustible/hot work controls), 2) rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic/manual fire suppression, fire brigade/pre-fire plans), and 3) provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, success path remains free of fire damage, recovery actions).

The fire water system is not part of echelon 1 of the defense-in-depth concept because it is not involved with administrative procedures to prevent fire from occurring. The fire water system is part of echelons 2 and 3 of the defense-in-depth concept. The use of the fire protection water for this non-fire protection system water demand has been shown to not adversely impact the ability of the fire water system to provide required flow and pressure for the largest regulatory required sprinkler system/hose stream demand. Therefore, this use of fire protection water for non-fire protection system water demand has no adverse affect on the ability of the fire water system to perform its intended function of providing an adequate fire water supply. Therefore, echelons 2 and 3 of the defense in depth concept are maintained.

RAI FPE 07

Table B-1; (1) please confirm which "open items" are closed and which should be implementation items in Attachment S, and (2) identify Table B-1 elements that require procedure or documentation to be updated and justify their exclusion as implementation items in Attachment S.

Waterford 3 Response

1. Confirm which “open items” are closed and which should be implementation items in Attachment S.

   Table B-1, Open Item identified in VFDR 3.9.1(1) has been closed.

   Variances from deterministic requirements (VFDR’s) identified in Table B-1 are categorized as “Open Items” or “Confirmatory Items”. All confirmatory items are identified in Attachment S as “Implementation Items” or “Plant Modifications”. Confirmatory items are identified as

implementation items because these items are solely required for NFPA 805 implementation. These items are not required for the current Appendix R licensing basis. Open items are required by the current Waterford 3 licensing basis, as well as NFPA 805, and must be corrected regardless of transitioning to NFPA 805. These open items are tracked by the plants corrective action program. Although Waterford 3 considers these open items as required actions to transfer to NFPA 805, they do not rise to the level of being categorized as confirmatory items needing to be listed in Attachment S.

While responding to this RAI, Waterford 3 Staff identified two Attachment S implementation items (S2-3 and S2-4) that were incorrectly categorized as "Confirmatory Items". These two items are required for compliance with the current Appendix R licensing basis and should have been categorized as "Open Items". Both of these items are tracked by the plants corrective action program, as correctly identified in Attachment A, Table B-1, Section 3.3.7.1, VFDR's 3.3.7.1-1(2) & (4) and 3.3.7.1-3. Therefore, these two items are to be considered "Open Items". However, implementation items S2-3 and S2-4 will remain in Attachment S.

2. Identify Table B-1 elements that require procedure or documentation to be updated and justify their exclusion as implementation items in Attachment S.

   Current deviations to Table B-1 elements that require procedure or documentation to be updated are identified as "open items" or "confirmatory items". Changes to all procedures and documents relative to implementation of NFPA 805, including those identified in Table B-1 and Attachment S, are regulated by the plant configuration control process, which includes the use of approved procedures (i.e. EN-DC-115, EN-DC-126, EN-DC-128, EN-LI-100, etc). Attachment S Implementation Item S2-11 ensures required procedure and engineering documents are developed/updated prior to transition to NFPA 805.

RAI FPE 08

Attachment L, Approval Request 1 -A deviation from NFPA 51 B, "Standard for Fire Prevention During Welding, Cutting, and Other Hot Work," is requested, as referenced in NFPA 805 Section 3.3.1.3.1, to allow for hot work to be performed in sprinklered buildings, while such protection system is impaired/disabled. Please provide additional information to include: any limits or controls in place during these scenarios, identify which fire areas are expected to endure hot work with a disabled sprinkler system, and what compensatory measures will be established to provide assurance of meeting the nuclear safety performance criteria.

Waterford 3 Response

Limits or controls in place during these scenarios:

The following limits and controls are administratively controlled by Procedure EN-DC-127, Rev. 11 "Control of Hot Work and Ignition Sources"

1. Written approval and guidance is required from the Fire Protection Engineer or Designee prior to commencing work.
2. Operations are notified prior to the initiation of all Hot Work. This notification is required once per shift.
3. A hot work fire watch with dedicated (not permanent) fire extinguisher (compatible with the environmental conditions encountered at the work site) is required to be present during the

hot work activity and remain in the immediate work area for a minimum of 30 minutes following completion of the hot work activity.

4. Combustible material within 35 feet of the work area (both horizontally and vertically if floor openings exist) that could become ignited from the Hot Work are: 1) Removed. 2) Protected by covering with metal guards or flame retardant fabric. Or 3) Protect the work area using metal guards or flame retardant fabric in order to prevent the spread of sparks, slag, and molten metal from the work area
5. Equipment is checked prior to the activity and is in good working condition (oxygen/acetylene hoses and tanks, regulators, backflow preventers welding leads, etc.) and is restrained properly.
6. Shut off valves at the compressed gas cylinders are closed, regulators are de-pressurized, and welding machines turned off when equipment is not in use.

Identify which fire areas are expected to endure hot work with a disabled sprinkler system:

Fire Areas/Fire Zones protected by sprinkler systems necessary for compliance with NFPA 805 are identified in LAR Table 4-3. Hot work activities in any of these areas could be postulated that would necessitate the need to disable the sprinkler system during the hot work activity.

What compensatory measures will be established to provide assurance of meeting the nuclear safety performance criteria:

Compensatory measures specified in TRM 3.7.10.2 for inoperable spray and/or sprinkler systems are established when the sprinkler system is disabled.

RAI FPE 09

Attachment L, Approval Request 5 -Two previously approved deviations from RCP oil collection requirements now found within NFPA 805 Section 3.3.1.2 are discussed. Both deviations, one regarding the reduced capacity of the oil collection tanks and another regarding remote oil fill lines not protected by the RCP oil collection system, relied on suppression and detection installed as part of the approvals. However, since then, the need for these suppression systems has been reevaluated (see ER-W3-2001-1174-000) and these suppression and detection systems were removed without NRC approval. These deviations without the suppression and detection have been submitted as part of the NFPA 805 transition. In addition, the licensee indicated that certain previously submitted information is now inaccurate due to the fact that these systems are no longer installed as previously stated (see W3P84-0709, W3F1-97-0021, W3F1-97-0117, W3F1-97-0191).

a. *Please provide further description summaries of the previous deviations and conditions. Including a summary of the originally approved suppression and detection configurations.*
b. *Please provide a summary description of the previous approval evaluation (e.g. ER-W3-2001-1174-000), including the basis for the conclusion on the removal of detection and suppression systems which was credited in the previously approved deviations.*
c. *Please provide a summary of the resolution of each deviation.*

Waterford 3 Response

The above RAI refers to the previously approved deviations from RCP oil collection requirements now found within NFPA 805 Section 3.3.1.2. This NFPA Section number is incorrect. NFPA 805 Section 3.3.12 is the NFPA Section dealing with RCP oil collection requirements.

a. Provide further description summaries of the previous deviations and conditions. Including a summary of the originally approved suppression and detection configurations:

   Deviation Regarding Reduced Capacity of the Oil Collection Tanks: One previous deviation approved by the NRC in Safety Evaluation Report, Supplement No. 8 dated December 1984 has existed since initial licensing of Waterford 3 and was a deviation to technical requirements of 10CFR50, Appendix R, Section III.O to provide an oil collection system capable of holding the entire lube oil system inventory from all Reactor Coolant Pumps. A Reactor Coolant Pump (RCP) Oil Collection System is provided for each pump to direct lube oil from pressurized and unpressurized leakage sites to a collection tank. The RCP Oil Collection System consists of oil drip pan/enclosures mounted on each reactor coolant pump motor. A gravity drain piping system transports any accumulated oil from the drip pan/enclosures to an oil collection tank. There are two 200 gallon oil collection tanks. One tank serves pumps RCP 1A and 1B and the other tank serves pumps RCP 2A and 2B. The tanks are located inside the Reactor Containment Building outside the biological shield wall at EL. -4.00 ft msl. Each tank is capable of collecting oil from one RCP oil lube system (195 gallons); vented and provided with a flame arrester; and furnished with a glass liquid level gauge to provide local indication of existence of oil in the tank. Each RCP Motor Lube Oil System has an alarm which will sound in the Control Room to alert operators if a significant amount of oil is lost from the lube oil reservoirs. The RCP Oil Collection System and Reactor Coolant Pumps Lube Oil System are seismically designed such that there is reasonable assurance that the system will withstand the safe shutdown earthquake. This is consistent with NRC memo; R. H. Vollmer to D. B. Eisenhut, dated April 1, 1983, Position 3. Thus very limited quantities of escaping oil are postulated. A fire postulated for that condition would be characterized by the rapid consumption of the oil if ignited by hot surfaces and would not pose a significant fire hazard. Review of Calculation ECF00-026 “Post Fire Safe Shutdown Analysis” reveals that the only safe shutdown component located in the vicinity of the RCPs is steam generator pressure indication. Should this pressure indication be lost due to fire conditions, there are pressure transmitters located outside of containment on the main steam lines that can be utilized for this indication. Therefore, safe shutdown would not be adversely affected should this indication be lost due to fire conditions. The potential for the catastrophic total release of the entire lube oil inventory of two RCPs is very remote and would generally constitute an incredible multiple system failure. If total release exceeds the collection tank capacity, the tank overflows and would drain to the containment sumps without contacting hot surfaces thus would not pose a significant fire hazard.

   Deviation Regarding Remote Oil Fill Lines not Protected by the RCP Oil Collection System: Two phases of plant modifications (Station Modification SM-1353 “Phase I/upper RCP reservoirs” and Design Change DC-3318 “Phase II/lower RCP reservoirs”) were initiated in 1986 and 1990 that installed remote RCP oil fill lines. These remote fill lines were installed to reduce dose to workers who periodically add oil to the RCP lube oil systems during power operation. The engineering evaluation performed, prior to implementation of the modification, technically justified not providing an oil collection system for these lines and determined that the modification met the intent of 10CFR50, Appendix R, Section III.O. However, results of later reviews by Waterford 3 plant staff identified that, while the design meets the intent of Appendix R, it was not in verbatim compliance with Appendix R.

Subsequently, Waterford 3 submitted for and received NRC approval of this deviation as identified in Waterford 3 LAR, Attachment L, Approval Request 5.

A RCP Oil Collection System is designed and installed as described above in "Deviation Regarding Reduced Capacity of the Oil Collection Tanks". Installation of the remote RCP oil fill lines provided a means to safely add oil to the upper and lower reservoirs on the RCP Motors from outside the D-Ring during all modes of operation. The modification consisted of routing 1 inch diameter stainless steel tubing from the existing fill connections on the motors up the inside of the D-Rings, through the feedwater piping penetrations in the shield wall and terminating adjacent to an installed hand pump. Flexible stainless steel hose is provided at the connection to the RCP motor to allow for thermal movement and vibration. The existing RCP Oil Collection System will collect any leakage or overflow at the remote fill line connection to the RCP Motor and route it to the applicable Oil Collection Tank located outside the D-Ring. The remote fill line tubing is non-safety class and seismically supported. Compression type tube fittings are used and a functional leak test was performed after installation to check for leaks.

The remote oil fill lines do not contain standing oil and are infrequently used on an as needed basis. The remote oil fill lines do not contain standing oil based on the following: a) the existing RCP oil reservoir configurations include overfill drain and vent lines situated below the entry points of the remote fill lines at the RCP; these overfill provisions route excess oil to the oil collection system, and b) the remote oil fill lines are capped when not in use, such that no vent path exist in the fill lines. This will cause any oil which might be forced from the reservoirs to flow out of the overfill provisions and then be routed to the oil collection system instead of flowing back up the remote fill line tubing.

Transient combustibles for all areas, including the Reactor Containment Building, are controlled in accordance with Procedure EN-DC-161 "Control of Combustibles". The addition of oil to the RCP lube oil reservoirs is performed in accordance with Model Work Orders (previously reported in Waterford 3 Letter No. W3F1-97-0191 as Repetitive Task Work Authorization Packages). These Model Work Orders and plant maintenance work controls procedures ensure the following:

1. The proper amount of lube oil needed is added, thus reducing the likelihood of overfilling the RCP reservoirs. This is accomplished by checking the Upper and/or lower lube oil reservoir oil level(s), as applicable before determining the quantity of oil to be added.
2. Visual inspection of the floor level (-11 elevation) under the applicable RCP is performed after entry and before exit of the Containment Building to ensure appropriate actions are taken should a leak occur.
3. The oil collection tank is empty before fill activity personnel exit the Containment Building. This is to assure that the tank has room to hold the lube oil volume of a RCP.
4. After leaving Containment the expected rise in reservoir level occurred as a result of the oil addition is verified.
5. If oil level increase is not representative of the amount of oil added, System Engineering is notified to determine if there was an oil leak involved

Postulating an oil spill of the maximum credible oil addition quantity for the floor area of a single D-Ring represents a fire severity of less than 2 minutes. Recognizing that for this scenario to occur, oil addition operations would be in progress and personnel would be inside the RCB. It is reasonable to believe that the personnel adding oil would notice an oil spillage of this magnitude while checking for oil below the applicable RCP (per the MWO/PMID) on their way out of the Containment Building. Response would be immediate

and manual fire fighting actions would be expeditious. This, accompanied by the fuel package limitations of less than 2 minute severity, indicates that damage would be minimal. As indicated above the only safe shutdown component located in the vicinity of the RCPs is steam generator pressure indication. Should this pressure indication be lost due to fire conditions, there are pressure transmitters located outside of Containment on the main steam lines that can be utilized for this indication. Therefore, safe shutdown would not be adversely affected.

Summary of the Originally Approved Suppression and Detection Configurations: At the time of the above described NRC deviation approval, the RCP's were protected by multicycle pre-action sprinkler systems actuated by a line-type thermal detection system (Thermistor-Wire by Allison Controls, Inc). The Allison Controls, Inc. line-type thermistor-wire fire detection system was replaced in the mid 1990's by a Cerberus Pyrotronics fire detection system consisting of spot-type thermal detectors (Ref. Modification DCP3268). These spot-type thermal detectors were strategically located around the RCP motors at approximately the +46 feet and +21 feet elevations. The multicycle pre-action sprinkler system was converted to a manually actuated pre-action sprinkler system at the same time with manual actuation pull stations located in the Control Room and at the sprinkler system control valves. The sprinkler systems consisted of four rings of directional sprinkler nozzles located at strategic elevations around each RCP. Sprinklers protecting RCP's 1A and 1B were supplied by Sprinkler System FPM-1 and sprinklers protecting RCP's 2A and 2B were supplied by Sprinkler System FPM-2.

b. Provide a summary description of the previous approval evaluation (e.g. ER-W3-2001-1174-000), including the basis for the conclusion on the removal of the detection and suppression systems which was credited in the previously approved deviations:

The suppression systems were originally installed to meet insurance requirements and their unavailability does not impact the ability of the oil collection systems to perform their intended function. Inoperable/deleted RCP suppression systems do not significantly impact previously accepted deviations from 10CFR50, Appendix R, Section III.O. Therefore the reference to the suppression systems in approval of the deviation is not material to the Staff's original basis for approval. The following basis was provided in ER-W3-2001-1174-000:

1. The design of the Waterford 3 RCP lube oil collection system is such that it forms a complete enclosure over potential leakage points (with the exception of the remote oil fill lines), thus effectively eliminating the possibility of any oil escaping from the confines of the collection system.
2. The remote oil fill line is a normally empty line. Oil addition activities are administratively controlled such that, in the unlikely event of a fire, the response to the fire would be rapid and conclusive.
3. The majority of hot RCP piping is insulated, thus further isolating potential ignition sources from any escaping lube oil.
4. The lube oil collection system and the lube oil system are both seismically designed to preclude their failure during a Safe Shutdown Earthquake. Thus very limited quantities of escaping oil are postulated. A fire postulated for that condition would be characterized by the rapid consumption of the oil, if ignited by hot surfaces.
5. Industry experience, that was not available at the time of the original deviation request (including NRC Information Notice 94-58), indicates that RCP lube oil fires are typically precluded by alarms associated with bearing temperatures and low oil levels. In addition, according to this data, these fires can be attributed to deficiencies in the design of the lube oil collection system. The Waterford 3 RCP Lube Oil Collection

System was specifically field inspected by the NRC Staff during an inspection conducted April 6 through May 17, 1997. Inspection Report 50-382/97-08 documents the NRC Staff conclusion that "…The RCP lube oil drain and fill systems were well installed and maintained.

6. The potential for the catastrophic total release of the entire lube oil inventory of two RCPs is very remote and would generally constitute an incredible multiple system failure. The existing lube oil tanks are sized for the inventory of one RCP.

Based on the above, not crediting the suppression and detection systems for the subject deviations do not significantly impact the deviations.

c. Provide a summary of the resolution of each deviation:

Deviation Regarding Reduced Capacity of the Oil Collection Tanks: A RCP Oil Collection System is provided for each RCP to direct lube oil from pressurized and unpressurized leakage sites (with the exception of remote oil fill lines) to a collection tank. There are two collection tanks, each collection tank serves two RCPs and each collection tank has sufficient capacity to contain the entire quantity of oil in one RCP. The lube oil collection system and the lube oil system are both seismically designed to preclude their failure during a Safe Shutdown Earthquake. Thus very limited quantities of escaping oil are postulated. A fire postulated for that condition would be characterized by the rapid consumption of the oil if ignited by hot surfaces and would not pose a significant fire hazard. The only safe shutdown component located in the vicinity of the RCPs is steam generator pressure indication. Should this pressure indication be lost due to fire conditions, there are pressure transmitters located outside of containment on the main steam lines that can be utilized for this indication. Therefore, safe shutdown would not be adversely affected. The potential for the catastrophic total release of the entire lube oil inventory of two RCPs is very remote and would generally constitute an incredible multiple system failure. If total release exceeds the collection tank capacity, the tank overflows and would drain to the containment sumps without contacting hot surfaces thus would not pose a significant fire hazard.

Deviation Regarding Remote Oil Fill Lines not Protected by the RCP Oil Collection System: The design of the Waterford 3 RCP lube oil collection system is such that it forms a complete enclosure over potential leakage points (with the exception of the remote oil fill lines), thus effectively eliminating the possibility of any oil escaping from the confines of the collection system. The existing RCP Oil Collection System will collect any leakage or overflow at the remote fill line connection to the RCP Motor and route it to the applicable Oil Collection Tank located outside the D-Ring. The remote fill line tubing is seismically supported and does not contain standing oil except when fill operations are in process. These fill operations are infrequent and are used on an as-needed-basis. Work control procedures control the addition of lube oil to the RCPs to ensure that only the proper amount of lube oil needed is added, the oil being added is going to the proper reservoir in the RCP motor, visual inspection of the floor level below the applicable RCP is performed at entry and exit of the Containment Building to ensure appropriate actions are taken should a leak occur, and the oil collection tank is empty before fill activity personnel exit the Containment Building.

Postulating an oil spill of the maximum credible oil addition quantity for the floor area of a single D-Ring represents a fire severity of less than 2 minutes. Recognizing that for this scenario to occur, oil addition operations would be in progress and personnel would be inside the RCB. It is reasonable to believe that the personnel adding oil would notice an oil spillage of this magnitude while checking for oil below the applicable RCP on their way out of the Containment Building. Response would be immediate and manual fire fighting actions would be expeditious. This, accompanied by the fuel package limitations of less than 2 minute severity, indicates that damage would be minimal. The only safe shutdown

component located in the vicinity of the RCPs is steam generator pressure indication. Should this pressure indication be lost due to fire conditions, there is pressure transmitters located outside of containment on the main steam lines that can be utilized for this indication. Therefore, safe shutdown would not be adversely affected.

RAI FPE 10

Attachment L, Approval Request 6 -A deviation from NFPA 805 Section 3.5.1 is requested for several plant areas where the fire water supply demand is not met with the least demanding portion of the main loop out of service. Please justify this deviation to include a discussion of the following:

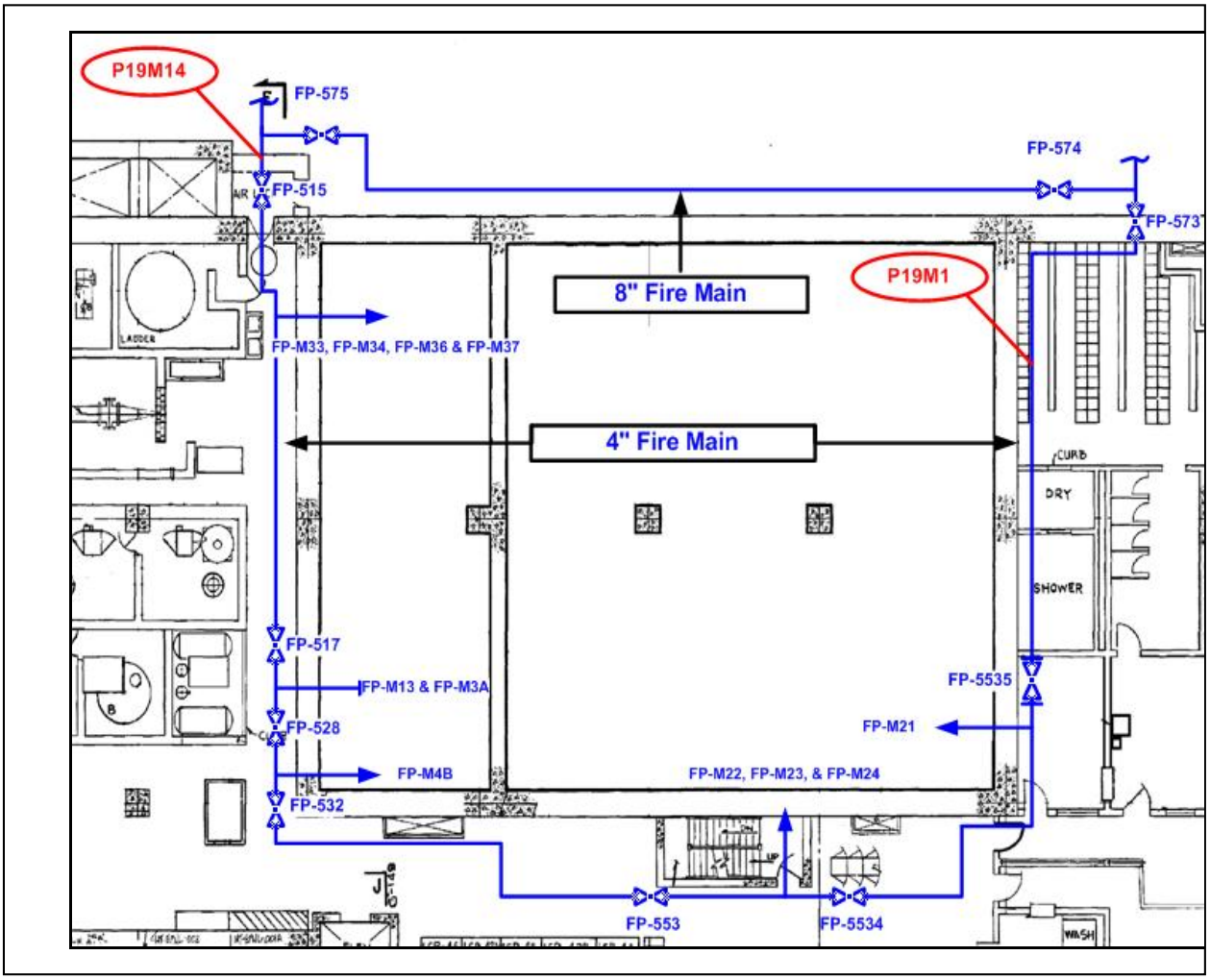
a. *The location of this 4" loop section.*
b. *The procedures in place to mitigate.*
c. *The capacity of the three sprinkler systems given the least demanding portion of the main loop is out of service.*

Waterford 3 Response

Additional information as requested above is provided in support of Attachment L, Approval Request 6.

a. Location of this 4" loop section:

An 8 inch diameter fire main is routed through the Reactor Auxiliary Building (RAB) and connects to the outside 10 inch diameter underground fire main loop on both the east and west sides of the RAB. This 8 inch fire main traverses east/west near the south wall of Fire Area RAB 32 (RAB -4' Elevation). The 4 inch fire main loop, addressed in Attachment L Approval Request 6, connects to the above described 8 inch fire main in Fire Area RAB 32 (approximately 3 feet west of column line 10AZ and approximately 2'-8" north of column line L). The 4 inch loop then traverses south into Fire Area RAB 30, then west and north in Fire Area RAB 31 around the Condensate Storage and Refueling Water Pools before re-entering Fire Area RAB 32 (RAB -4'Elevation) over fire door D-161 (along column line L between column lines 5A and 6A). The 4 inch loop then connects back into the 8 inch fire main loop in RAB 32 (approximately 7'-2" west of column line 6A and approximately 11'-6" north of column line L). This 4 inch loop is approximately 300 linear feet of pipe in length and runs in the overhead areas of the -4 foot elevation in the RAB. See Sketch below for general routing of the 8 and 4 inch fire mains discussed above.



SKETCH (Not to Scale)
RAB -4' Elevation

b. <u>Procedures in place to mitigate</u>:

Procedure FP-001-015 "Fire Protection System Impairments", Section 5.2.3 specifically addresses the impact of closing sectional isolation valves on the 4 inch fire main loop described above. This procedure identifies which sprinkler systems are impaired based on which individual fire main sectional isolation valve (FP-515, FP-517, FP-528, FP-532, FP-553, FP-5534, FP-5535, or FP-573) on the 4 inch fire main loop is closed. For each sprinkler system impaired, compensatory measures specified in TRM 3.7.10.2 are implemented during the time the sprinkler system is impaired.

c. <u>Capacity of the three sprinkler systems given the least demanding portion of the main loop is out of service</u>:

System design and water demand requirements for Sprinkler Systems FP-M3A, FP-M4B, and FP-M22 are summarized below in Table 1. These water demand requirements include sprinkler demand as well as 500 gpm demand for fire hose. With the least demanding portion of the 4 inch fire main loop out of service, the three sprinkler systems retain a minimum capacity of approximately 70%. This minimum capacity is expressed as a percentage of the sprinkler system design demand area (shown in Table 1) that can be supplied at the system design density (also shown in Table 1). This percentage of the system design demand area represents the most hydraulically remote portion of the sprinkler system coverage area.

Table 1

Sprinkler System No.	**Sprinkler System Design Demand**	**Sprinkler Demand at Base of Riser**
FP-M3A	0.25 gpm/sq.ft. over the entire room area (approximately 2000 sq. ft.)	592 gpm at 45 psi
FP0M4B	0.25 gpm/sq.ft. over the entire room area (approximately 2000 sq. ft.)	676.5 gpm at 70.2 psi
FP-M22	0.16 gpm/sq.ft. over the most remote 1500 sq. ft.	730.7 gpm at 73.3 psi

Summary:

As demonstrated above, the fire water supply system at Waterford 3 is adequate to supply a sizable portion of the system design demand for Sprinkler Systems FP-M3A, FP-M4B, or FP-M22 with the least demanding portion of the 4 inch fire main loop out of service. Sectional isolation valves are provided such that Sprinkler System FP-M3A, FP-M4B, or FP-0M22 can be isolated without impairing inside hose station coverage for the area of the impaired sprinkler system. Therefore, inside hose stations are available for fire brigade use during periods when sprinkler system is being partially or completely impaired. Procedural controls in Fire Protection System Impairments Procedure FP-001-015 ensure TRM required compensatory measures for inoperable suppression system are in place when the least demanding portion of the 4 inch fire main loop is out of service.

The above described features ensure adequate compensatory measures are implemented for Sprinkler System FP-M3A, FP-M4B, and/or FP-M22 when the fire water supply demand can not be met due to the least demanding portion of the 4 inch fire main loop being out of service. This ensures the defense-in-depth concept with regard to fire protection is maintained.

RAI FPE 11

LAR, Section 4.1.3 (Power Block) includes various example structures including "service building" and "intake structure" when referring to NEI 04-02. In addition, FAQ-19, FP-101-018 and LAR, Attachment E (pages E-4 through E-9) mention several other structures or fire areas not currently listed under LAR, Attachment I. Please justify the exclusion of the following structures and/or fire areas from the power block definition (LAR, Attachment I):

a. *DS -Discharge Structure*
b. *Intake Structure and associated ductbanks*
c. *LLRWSF -Low Level Radwaste Storage Bldg*
d. *CP -Condensate Polisher Bldg*
e. *RMSB -Radwaste Material Storage Bldg*
f. *RSB -Radwaste Solidification Bldg*

g. SB -Service Building

h. WTB -Water Treatment Bldg

i. CHLR -Chiller Bldg

Waterford 3 Response

All structures identified above do NOT meet the radiological release and nuclear safety performance criteria described in Section 1.5 of NFPA-805, and are therefore excluded from the referenced Power Block definition. Details are provided below:

The Discharge and Intake Structures and associated ductbanks were excluded as the structures are not required to support any nuclear safety performance criteria as defined in Section 1.5 of NFPA-805, which includes 1.) Reactivity Control, 2.) Inventory and Pressure Control 3.) Decay Heat Removal 4.) Vital Auxiliaries and 5.) Process Monitoring. The discharge and intake structures are required for reliable production of electricity but are not required for the safe shutdown of the plant.

For the LLRWSF, the Low Level Radwaste Storage Bldg was excluded because this building is located outside the protected area and not connected to the plant. This building does not contain components required for safe shutdown.

The Condensate Polisher building is used to process the full condensate flow and to produce effluent meeting required specifications. The system is not essential for nuclear plant operation and may be taken out of service without limiting plant operations.

The Radwaste Material Storage Building and Radwaste Solidification Building house low level radioactive waste and equipment. Neither structure contains components required for safe shutdown.

The Service Building provides office space and shop area to maintenance and operations personnel. The Service Building also contains a warehouse. This building does not contain components required for safe shutdown.

The Water Treatment and Chiller Buildings are used to process “raw” water and provide chilled water to non-safety related air handling units located in the Reactor Auxiliary Building during normal plant operation. Neither structure contains components required for safe shutdown or nuclear plant operations.

In every case, the structures are not required to meet a nuclear safety goal and do not exceed radiological release per 10CFR20 Part 20 limits.

RAI FPE 12

Attachment L, Approval Request 2 -A deviation from NFPA 805 Section 3.3.5.1 is requested for low voltage video/communication/data cables located above suspended ceilings which are not listed for plenum use, routed in armored cable, routed in metallic conduit, or routed in cable trays. Please provide further clarification on: which fire areas contain these cables, whether or not there is detection located above the ceiling in these areas, and provide any design

specifications that will ensure any future cable installation will meet NFPA 805 Section 3.3.5.1. In addition, describe any administrative controls in place to limit combustible material in vicinity of these cables.

Waterford 3 Response

Which fire areas contain these cables and there is detection located above the ceiling in these areas:

This information is provided in the following table:

Building / Elevation	**Fire Area (Fire Zone)**	**Area Description / Portion with Suspended Ceilings**	**Detection Provided Above Suspended Ceiling (Yes/No)**
RAB -4	RAB 30	Administration Area / East of 10A & North of H, except for Storage Room located north & west of elevator and Janitor's Closet located north of J & east of 11A.	No (Except detection is provided above ceiling in Chemistry Office.
RAB +7	RAB 27	Mechanical-Electrical HVAC Equipment & Administrative Envelope / All areas except HVAC Room, Multiplexer Room, Pipe Chase, I & C Room (south of acoustical folding partition), and Communication Room.	Yes
RAB +46	RAB 1 (RAB 1A)	Main Control Room Proper / All	No
	RAB 1 (RAB 1C)	Control Room Emergency Living Quarters / All	No
	RAB 1 (RAB 1D)	Computer Room / All	No

Provide any design specifications that will ensure any future cable installation will meet NFPA 805 Section 3.3.5.1:

Engineering Standard EN-IC-S-002-W "Communication Cable and Equipment Installation" will ensure future cable installation will meet NFPA 805 Section 3.3.5.1. LAR Attachment S, Implementation Item S2-11 ensures required engineering standards and other engineering documents are developed/updated to ensure future installation of these cables are in compliance with NFPA 805 Section 3.3.5.1.

Describe any administrative controls in place to limit combustible material in vicinity of these cables:

Engineering Standard EN-IC-S-002-W "Communication Cable and Equipment Installation" controls installation of permanent combustible materials in these areas. EN-DC-161 "Control of Combustibles" controls transient combustibles in all areas. Although these procedures and standards do not specifically limit combustible materials in the vicinity of cabling installed above suspended ceilings, they do ensure combustibles are controlled in compliance with the plant's approved fire protection program.

RAI FPE 13

Attachment L, Approval Request 4 -A deviation from NFPA 805 Section 3.3.5.3 is requested for use of limited quantities of video/communication/data cabling not meeting flame propagation tests acceptable to the NRC. During the audit, a communication room on +7 EL. was observed with large quantities of data/communication cabling that do not appear to have acceptable flame propagation ratings (see FAQ-06-022). This room currently contains fire detection but no suppression. Please clarify whether or not all of these cables meet the flame propagation ratings acceptable to the NRC. For those cables without acceptable ratings, Please provide further justification and details for why these cables do not need to meet NFPA 805, Section

3.3.5.3. Include any fire load calculations and/or protective measures. In addition, provide further clarification on which fire areas contain these cables, any design specifications that will ensure any future cable installation/replacement will meet NFPA 805, Section 3.3.5.3, and describe any administrative controls to limit combustible material in vicinity of these cables.

Waterford 3 Response

Communication Room on +7 EL: Clarify whether or not all of these cables meet the flame propagation ratings acceptable to the NRC:

Exposed cabling in the Communication Room is a combination of cables that meet the flame propagation ratings acceptable to the NRC and those that do not. Approximately half of the electrical cables in the Communications Room meet acceptable flame propagation ratings.

Communication Room on +7 EL: For those cables without acceptable ratings, provide further justification and details for why these cables do not need to meet NFPA 805, Section 3.3.5.3. Include any fire load calculations and/or protective measures:

The communications room, which contains a quantity of exposed cabling, represents a moderate fire loading (less than 2-hour fire duration) when compared to the floor area of the room. The room has automatic fire detection, manual fire hose and fire extinguisher coverage and is separated from other fire areas by 3-hour rated barriers and from other rooms within Fire Area RAB-27 by substantial concrete masonry walls and metal door. A fire in the communications room would be detected rapidly by the fire detection system and the present fire hazard is well within the capability of the plant fire brigade to control and extinguish. The Communication Room contains no equipment required for safe shutdown. The substantial concrete masonry walls and metal door separating the communications room from other rooms within Fire Area RAB-27 is considered adequate to prevent fire propagation outside the room. However, if fire should propagate across this barrier, there are insufficient combustibles in surrounding areas to propagate the fire to damage redundant equipment required for safe shutdown. Therefore, these cables (without acceptable ratings) are considered an acceptable fire hazard and do not adversely affect safe shutdown.

Provide further clarification on which fire areas contain these cables:

Field walk-down verified these cables are in the following fire areas: Fire Area RAB 1, Fire Zone RAB 1A “Main Control Room Proper”, Fire Zone RAB 1C “Control Room Emergency Living Quarters”, Fire Zone RAB 1D “Computer Room”; Fire Area RAB 27 “Mechanical-Electrical HVAC Equipment & Administrative Envelope”; and Fire Area RAB 30 “Administration Area (Health Physics). These areas contain a limited quantity of this wiring which is sufficiently dispersed, considered an insignificant fire hazard, and not capable of causing fire damage to components necessary for safe shutdown. Combustibles associated with these cables are

considered insignificant with regards to combustible loading in the affected areas due to the limited quantity of these cables.

Design specifications that will ensure any future cable installation/replacement will meet NFPA 805, Section 3.3.5.3:

Engineering Standard EN-IC-S-002-W “Communication Cable and Equipment Installation” will ensure future cable installation will meet NFPA 805 Section 3.3.5.1. LAR Attachment S, Implementation Item S2-11 ensures required engineering standards and other engineering documents are developed/updated to ensure future installation of these cables are in compliance with NFPA 805 Section 3.3.5.1.

Describe any administrative controls to limit combustible material in vicinity of these cables:

Engineering Standard EN-IC-S-002-W “Communication Cable and Equipment Installation” controls installation of permanent combustible materials in these areas. EN-DC-161 “Control of Combustibles” controls transient combustibles in all areas. Although these procedures/standards do not specifically limit combustible materials in the vicinity of cabling installed above suspended ceilings, they do ensure combustibles are controlled in compliance with the plants approved fire protection program.

RADIOACTIVE RELEASE (RR)