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Waterford 3

W3F1-2014-0025

June 11, 2014

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

SUBJECT: Updated Responses to Request for Additional Information Regarding Adoption of National Fire Protection Association Standard NFPA 805 License Amendment Request (LAR) Waterford Steam Electric Station, Unit 3 (Waterford 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. Entergy letter W3F1-2012-0064 "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" dated September 27, 2012
4. Entergy letter W3F1-2012-0083 "90 Day 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" dated October 16, 2012
5. Entergy letter W3F1-2013-0022 "Response to 2nd Round Request for Additional Information Regarding Adoption of National Fire Protection Association Standard NFPA 805 License Amendment Request, Waterford Steam Electric Station, Unit 3" dated May 16, 2013
6. Entergy letter W3F1-2013-0048 " Supplement to NFPA 805 License Amendment Request (LAR) Waterford Steam Electric Station, Unit 3" dated December 18, 2013
7. NRC Transmittal to Entergy dated March 9, 2012,"Request to Reinstate Enforcement Discretion RE: License Amendment Request to Transition to National Fire Protection Association Standard 805 (TAC NO. ME7602)

Dear Sir or Madam:

By letter dated November 17, 2011, as supplemented by letters dated January 26, September 27, October 16, 2012, May 16, 2013, and December 18, 2013 (References 1 through 6 respectively), Entergy Operations, Inc. (Entergy), submitted a license amendment request (LAR) to transition its fire protection license basis at the Waterford Steam Electric Station, Unit 3, from paragraph 50.48(b) of Title 10 of the *Code of Federal Regulations* (10 CFR) to 10 CFR 50.48(c), "National Fire Protection Association Standard 805" (NFPA 805).

The LAR Supplement provided in Reference 6 represents changes to specified LAR Attachments and supporting calculations primarily as a result of performing extensive reanalysis utilizing only NRC-accepted methods. To assist the NRC in reviewing the LAR Supplement provided in Reference 6 and the supporting documentation, Entergy presented the summary of changes during a meeting held on February 4, 2014. In response to feedback provided by the NRC during the presentation and several subsequent telephone meetings, to simplify the licensing basis, Entergy is superseding all previous RAI responses (References 3, 4 and 5) with the updated responses contained in Enclosure 2.

The RAI responses provided herein reside in one of three categories: Unchanged, Revised, or Superseded. Below is a brief description of each category.

<u>Unchanged</u> – These RAI responses are not impacted by the recent reanalysis and LAR Supplement. The responses come from the prior submittals and are included for completeness.

<u>Revised</u> – These RAI responses have been edited or replaced to address the requested information within the context of the LAR Supplement and supporting analysis. In many cases, the methodologies presented in the original responses remain valid; however, the results or referenced documents have changed. Revision bars are included to highlight the changed text.

<u>Superseded</u> –The previously provided RAI responses are withdrawn (References 3, 4, and 5). A brief explanation is provided to explain why the RAI request is no longer applicable. The majority of these meet one of two conditions. Either the cited situation being questioned no longer exists or the RAI requests justification of a methodology used to support the original LAR (Reference 1), which has since been replaced by an accepted methodology supporting the LAR Supplement (Reference 6).

Enclosure 1 contains the index of all previously submitted RAI responses divided in the categories discussed above. Please note that due to the nature of how RAIs are written and presented to the licensee, any particular RAI may contain many subparts, which are individually categorized. Special notations accompany these responses for clarity. Enclosure 2 contains the responses for all RAIs, grouped by RAI topic (PRA-xx, SSA-yy, FPE-zz, etc).

The No Significant Hazards evaluation contained in Reference 1 is unchanged by the attached Enclosures and the LAR Supplement (Reference 6).

The NRC reinstated Enforcement Discretion based on the acceptability of the LAR package and Unreviewed Analysis Methods (UAM) sensitivity analyses (Reference 7). As stated above, the analyses supporting the LAR Supplement (Reference 6) utilizes only NRC-accepted methods. Since the reinstatement of Enforcement Discretion was based in part on justification for applying UAMs, which are no longer used, Entergy requests confirmation that the status of Enforcement Discretion remains unchanged.

Lastly, Entergy believes that the RAI responses contained in Enclosure 2, in conjunction with the LAR Supplement (Reference 6) and its supporting analyses address the concerns the NRC had with the methodologies used and presented in the original LAR (Reference 1). The information provided in Enclosures 1 and 2 are a focused presentation of information previously submitted to the NRC which should assist in timely review of the NFPA 805 LAR. Entergy welcomes the opportunity to meet, discuss and address any remaining concerns the NRC may have in completion of the safety evaluation.

There are no new regulatory commitments contained in this submittal.

If you require additional information, please contact the Regulatory Assurance Manager, John Jarrell at 504-739-6685.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 11, 2014.

Sincerely,

Michael R. Chisum

MRC/ajh

Enclosures:

Enclosure 1: Index of RAIs
Enclosure 2: RAI Responses

cc:	Marc L. Dapas Regional Administrator U. S. Nuclear Regulatory Commission Region IV 1600 E. Lamar Blvd. Arlington, TX 76011-4511	RidsRgn4MailCenter@nrc.gov
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Enclosure 1 to

W3F1-2014-0025

Index of RAIs
Waterford 3 NFPA 805 License Amendment Request

Index of RAIs
Waterford 3 NFPA 805 License Amendment Request

Additional information was requested by the NRC Staff on July 18, 2012 and March 22, 2013 in support of the Review for Waterford Steam Electric Station, Unit 3 (Waterford 3) License Amendment Request (LAR) dated November 17, 2011. Entergy provided LAR Supplemental information on December 18, 2013. The content of the LAR Supplement affects the responses to the requested information. The following lists indicate how each RAI is affected and presented in Enclosure 2.

Response Unchanged

PRA09	SSA01.01(2b,2c)	FPE01
PRA14	SSA02.01(b)	FPE01.01
PRA24	SSA03.01	FPE02
PRA41	SSA11	FPE03
PRA43(e,f)	SSA15	FPE04
PRA44 (d,g,i,m)	RR01	FPE06
PRA54(c)	RR02	FPE07
PRA55	RR03	FPE09
FM06	RR04	FPE10
MP01	RR05	FPE11
PROG01	RR06	FPE12
PROG02		
PROG03		

Response Revised

PRA01	PRA19	FM01 (a to i,m,p)	SSA01(b)
PRA02	PRA21	FM01.01	SSA01.01 (1a,1b,2a)
PRA03	PRA31	FM02	SSA02
PRA04	PRA38	FM02.01	SSA02.01 (a,c)
PRA05	PRA39	FM03(c)	SSA08
PRA06	PRA44 (a,h,k,l)	FM05	SSA10
PRA07	PRA45 (a,b,d,g,h)		SSA10.01
PRA08	PRA48		SSA12
PRA10	PRA49		FPE05
PRA11	PRA52		FPE08
PRA12	PRA53		FPE13
PRA13	PRA54(b,d)		FPE15
PRA15	PRA57(c)		
PRA17	PRA59(a,b)		

Request Superseded

PRA16	PRA27	PRA35	PRA45(c,e,f)	PRA57(a,b)	SSA01(a)	SSA09
PRA18	PRA28	PRA36	PRA46	PRA58	SSA03	SSA13
PRA20	PRA29	PRA37	PRA47	PRA59(c,d,e)	SSA04	SSA14
PRA22	PRA30	PRA40	PRA50	FM01(j,k,l,n,o)	SSA05	FPE13.01
PRA23	PRA32	PRA42	PRA51	FM03(a,b)	SSA06	FPE14
PRA25	PRA33	PRA43(a,b,c,d)	PRA54(a)	FM04	SSA07	
PRA26	PRA34	PRA44(b,c,e,f,j)	PRA56	FM07	SSA08.01	

Enclosure 2 to

W3F1-2014-0025

RAI Responses
Waterford 3 NFPA 805 License Amendment Request

RAI Responses
Waterford 3 NFPA 805 License Amendment Request

Additional information was requested by the NRC Staff on July 18, 2012 and March 22, 2013 in support of the Review for Waterford Steam Electric Station, Unit 3 (Waterford 3) License Amendment Request (LAR) dated November 17, 2011. Entergy provided LAR Supplemental information on December 18, 2013. The content of the LAR Supplement affects the responses to many of the previously submitted RAI responses. The following responses supersede all previous RAI response submittals. Note that revision bars in the right margin indicate revised portions and also that the "Unchanged" responses are included for completeness.

PROBABILISTIC RISK ASSESSMENT (PRA)

RAI PRA 01

Please describe how the evaluation includes the possible increase in heat release rate (HRR) caused by the spread of a fire from the ignition source to other combustibles and summarize how suppression is included in the evaluation.

Waterford 3 Response

No secondary ignition concerns (except for cables/targets in direct contact with the ignition source) were identified in the baseline Waterford 3 fire PRA that was used to support the original LAR submittal. The analysis has been revised to consider secondary ignition more thoroughly (results are included in LAR Supplement). In response to the RAI, walkdowns of Waterford 3 physical analysis units (PAUs) were performed and locations of plausible secondary ignition targets were recorded. Using insights from this activity and based on PAU specific assessments, secondary ignition targets were identified in only two PAUs which resulted in both the additional dependent faults and re-evaluation of the applied HRR based on the physical properties of the new targets/sources. This included the expansion of the fire zone of influence provided by the secondary ignition. These PAUs were RAB 8 and RAB 27.

Suppression is considered in the updated analysis consistent with the guidance in NUREG/CR-6850 and Supplement 1 to NUREG/CR-6850.

RAI PRA 02

Transient fires should at a minimum be placed in locations within the plant physical analysis units (PAUs) where conditional core damage probabilities (CCDPs) are highest for that PAU, (i.e., at pinch points). Pinch points include locations of redundant trains or the vicinity of other potentially risk-relevant equipment, including the cabling associated with each. Transient fires should be placed at all appropriate locations in a PAU where they can threaten pinch points. Hot work should be assumed to occur in locations where hot work is a possibility, even if improbable (but not impossible), keeping in mind the same philosophy. Please describe how transient and hot-work fires are distributed within the PAUs. In particular, please identify the criteria which determine where an ignition source is placed within the PAUs. Also, if there are areas within a PAU where no transient or hot-work fires are located since those areas are considered inaccessible, please describe the criteria used to define "inaccessible." Note that an inaccessible area is not the same as a location where fire is simply unlikely, even if highly improbable.

Waterford 3 Response

Transients (other than Hot Work)

The transient fire evaluations have been revised since the original Waterford 3 NFPA 805 LAR submittal. The updated transient assessments are documented in PRA-W3-05-006T. As part of the FPRA update, revised transient fire scenarios for all program-defined PAUs were completed. These transient fire analyses assume that transient fires are possible at any location within the individual PAUs and they do not discredit any unoccupied location within any of these selected PAUs. Occupied space is defined as floor area that is physically occupied by equipment or previously analyzed storage cabinets (e.g. - "electrical safety equipment storage" located in the switchgear rooms). Occupied floor area transient fire ignition frequency was not lost; but rather the ignition frequency became denser in the unoccupied areas of the specific PAU, maintaining the same overall PAU transient fire ignition frequency. In addition, the transient fire analyses used the 98th percentile heat release rate for transient ignition sources contained in NUREG/CR-6850 to determine fire scenario impacts.

Sub-PAUs were utilized to analyze the PAU on a more refined basis. The use of sub-PAUs combined with allowing a transient fire to occur anywhere within the individual sub-PAUs ensures that all pinch points are identified. Sub-PAUs were formed based on the impact of pinch points within the PAU.

The transient fire scenarios also looked at "likelihood of storage". To understand where transients might be stored a thorough plant-specific walkdown was conducted and a review of the combustible control procedure (EN-DC-161 R6) was completed. This portion of the analysis allows for frequency to be increased in those areas deemed to be a "high likelihood of storage" area. Since the transient fire ignition frequency is fixed, ignition frequency was reduced equally from all floor area that was considered a "low likelihood of storage". No frequency was lost and no floor area was assigned a zero frequency during this process.

Hot work was addressed using two methods, the first includes the transient fires caused by hot work frequency (those transient fires not involving cable ignition) with the rest of the transient fires frequencies (scenario impacts as discussed above), and the second method evaluates cable fires caused by hot work separately. See the Hot Work section of this response for more details.

There is one fire area that is considered inaccessible. It is the ‘radioactive pipe chase’ RAB 23A. It is a sealed compartment that has several cable trays/conduits running through it. It is sealed by concrete block walls. Access to this area requires the floor plugs or part of the wall to be removed.

Transients Due to Hot Work

Additionally, an updated analysis (PRA-W3-05-014) was performed for transient fire scenarios for cable tray fires caused by hot work and re-quantified the CDF and LERF using whole PAU impacts (whole room burns). By analyzing these scenarios at a whole PAU level, hot work is assumed to be possible at any location within the various PAUs and is not discredited in any location within any of the PAUs.

The ignition frequency for fires due to transient hot work activity such as welding and cutting (listed as bin 5, 11, or 31 of NUREG/CR-6850, depending on plant location of the particular PAU) was taken from the Waterford 3 Plant Partitioning and Fire Ignition Frequency Development report (PRA-W3-05-001). To facilitate the assignment of influence factors and to better isolate the impact of hot work from general maintenance activities, the maintenance weighting factor has been subdivided into two components: mechanical/electrical maintenance and hot-work. The weighting factors developed in the Waterford 3 Plant Partitioning and Fire Ignition Frequency Development Notebook were developed using the methodologies contained in NUREG/CR-6850 and are appropriate for this updated analysis. See response to RAI PRA 03 for additional discussion on “weighting factors”.

RAI PRA 03

Please discuss the calculation of the frequencies of transient and hot-work fires. Please characterize the use of the influence factors for maintenance, occupancy, and storage, noting if the rating "3" is the most common, as it is intended to be representative of the "typical" weight for each influence factor. It is expected that the influence factor for each location bin associated with transient or hot-work fires will utilize a range of influence factors about the rating "3," including the maximum 10 (or 50 for maintenance) and, if appropriate, even the rating "0." Note that no PAU may have a combined weight of zero unless it is physically inaccessible, administrative controls notwithstanding. In assigning influence factor ratings, those factors for the Control/Auxiliary/Reactor Building are distinct from the Turbine Building; thus, the influence factor ratings for each location bin are to be viewed according to the bin itself.

If any influence factors outside of the values identified in Table 6-3 of NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," September 2005[1], have been used, please identify the values used, identify the PAUs that use these factors, and justify the assigned factor(s).

Waterford 3 Response

The Fire PRA did not use factors outside of those provided in NUREG/CR-6850. The influence factors applied to the Waterford 3 physical analysis units (PAUs) are documented in the Waterford 3 Fire PRA Plant Partitioning and Fire Ignition Frequency Development report. The factors are listed in Table 22 of the report. No exceptions to the NUREG/CR-6850 were applied.

In developing transient fire frequencies, weighting factors for several variables were determined. The weighting factors for occupancy, storage, and maintenance (divided into hot work and mechanical/electrical type activities) were developed. Application of the weighting was in line with the guidance provided in NUREG/CR-6850. A rating of “0” was used for only one PAU. Zero was listed for maintenance, occupancy and storage for the Radioactive Pipe Chase. This area is physically inaccessible. Access to the area requires knocking a hole in a concrete block wall. All other influence factors applied were 1, 3, 10, or 50. While the rating of “3” was not the most common value applied (“1” was the most common), the average factor for all inputs was approximately 2.81 (Table 22 of Waterford 3 Fire PRA Plant Partitioning and Fire Ignition Frequency Development report).

The numerical rankings developed for Waterford 3 were summed for each PAU and then normalized across all PAUs included in the generic plant locations that are the basis of the fire ignition frequencies provided by NUREG/CR-6850. These generic locations for transient fire ignition frequency allocation are:

- Containment (COP)
- Control / Auxiliary / Reactor Building (CAR)
- Turbine Building (TB)
- Plant Wide (PW)

Each activity influence rating was assigned for each category for each PAU. The influence values provide a means to establish a relative, progressive ranking of the physical analysis units that is used to distribute the averaged aggregated fire frequencies to each PAU. NUREG/CR-6850, Table 6-3, provides a framework for this assignment and includes suggested influence factor values. This guidance was used to develop the ranking criteria for this analysis. Table 22 of PRA-W3-05-001, Revision 1 identifies the influence factor rankings, definitions and values that are used for development of the PAU influence factors as adapted from NUREG/CR-6850

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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 under section 4.2.4. 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 process for evaluation of defense-in-depth (DID) began with these risk assessments (FREs). The associated fire area risk (CDF) and consequences (CCDP) were reviewed by a site DID expert panel to address general defense-in-depth echelon imbalances. Areas with high ignition frequencies were discussed as areas where additional fire brigade or fire prevention methods may be needed. Areas with high CCDPs were discussed for additional fire prevention elements. Beyond the risk assessment, a discussion of fire fighting strategies and operations impacts was performed to determine if additional DID methods could enhance fire brigade response.

This review is documented in WF3-FP-13-00004 (Waterford 3 Defense in Depth Report for NFPA 805). During the process, if deficiencies were discovered that did not allow the defense in depth goal to be met, a modification or programmatic change was considered, documented and can be found in Attachment C and Attachment S of the Waterford 3 LAR Supplement.

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 Supplement, with identified actions or changes to plant included in Attachment S of the LAR Supplement.

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 Supplement.

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) calculation (CN-TDA-10-2 Revision 1) performed in support of post-fire safe shutdown analysis shows that for a control room fire scenario with loss of offsite power (LOOP) approximately 11,896 gallons of charging flow is required to maintain

RCS inventory for 7 days. 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. LAR Supplement Implementation Item S2-21 ensures that the nitrogen accumulators for certain EFW and CCW valves can provide 24 hours of operation. The nitrogen accumulators are tested to ensure that they can provide at least 10 hours of pressure (tested via surveillance procedure STA-001-005). This change is to adjust the allowable leakage criteria to ensure 24 hours of valve functionality. The EFW flow control valves can also be operated locally. In summary, assuming instrument air were not available post-fire and that the nitrogen accumulators would be discharged after 24 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) calculation (CN-TDA-10-2 Revision 1) performed in support of post-fire safe shutdown analysis shows that for a control room fire scenario with loss of offsite power (LOOP) approximately 767,488 gallons of EFW flow is required in order for SG#1 and SG#2 to act as the primary heat removal components. 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, 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. These type actions and recovery of failed equipment are not credited in the Fire PRA.

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 24 hours (crediting implementation item S2-22 in LAR Supplement) to replenish the accumulators or to assume manual control of the valves. Since the valves fail 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, and 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

Section 10 of NUREG/CR-6850 states that a sensitivity analysis should be performed when using the fire ignition frequencies in the Supplement instead of the fire ignition frequencies provided in Table 6-1 of NUREG/CR-6850. Please provide the sensitivity analysis of the impact on using the Supplement 1 frequencies instead of the Table 6-1 frequencies on core damage frequency (CDF), large early release frequency (LERF), delta (Δ) CDF, and ΔLERF for all of those bins that are characterized by an alpha that is less than or equal to one. If the sensitivity *analysis indicates that the change in risk acceptance guidelines would be exceeded using the values in Table 6-1, provide justification for not meeting the guidelines.*

Waterford 3 Response

A detailed sensitivity evaluation was completed to examine the impact of using Supplement 1 frequencies instead of those in Table 6-1 of NUREG/CR 6850. The study applied the fire ignition frequencies from Table 6-1 for those scenarios whose bins are characterized by an alpha that is less than or equal to one.

Details of the sensitivity analysis are documented in report PRA-W3-05-015. Included in the report is a reexamination of top 50 CDF and LERF scenarios risk rankings. This analyzed if any dramatic change in risk ranking existed when changing the fire ignition frequencies. Overall, the changes in risk ranking were not overly significant; and no new risk insights can be made.
The analysis showed that one PAU (Turbine Generator Building (TGB)) did not meet thresholds set forth in NRC Regulatory Guide 1.174 (Section 2.2.4) for region III ΔCDF. Likewise one PAU (RAB 7) did not meet thresholds set forth in NRC Regulatory Guide 1.174 (Section 2.2.4) for Region III ΔLERF. Both of these PAUs would still fall within the Region II criterion. All other PAUs remain below Region III criterion.

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 3 Fire Probabilistic Risk Assessment (FPRA) Scenarios Report," should be included.

Waterford 3 Response

A detailed human error analysis associated with MCR abandonment was completed for the Waterford 3 FPRA. No screening value is applied. The Fire PRA supporting the original 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 to support the LAR Supplement and the assumed value (0.1) was found not to be bounding.

The potential for a fire outside the MCR (Fire area RAB-1) that leads to MCR abandonment was evaluated. No identified scenarios were found to be significant. All abandonment scenarios are associated with RAB-1 which includes the control room proper as well as the cable vault.

RAI PRA 08

Attachment W of the LAR provides the ΔCDF and ΔLERF for the variances from the deterministic requirements (VFDRs) for each of the fire areas, but the LAR does not describe either generically or specifically how ΔCDF and ΔLERF were calculated. Please describe the method(s) used to determine the changes in risk reported in the Tables in Attachment W. The description should include:

a. A summary of Probabilistic Risk Assessment (PRA) model additions or modifications needed to determine the reported changes in risk. If any of these model additions used data or methods not included in the FPRA Peer Review, describe the additions.

b. Identification of new operator actions (not including post MCR abandonment which are addressed elsewhere) that have been credited in the change in risk estimates. If such actions are credited, describe how instrument failure is addressed in the human reliability analysis (HRA).

Waterford 3 Response

Attachment W was revised (and resubmitted entirely as part of the Waterford 3 LAR Supplement). The revised Attachment W includes a discussion on the specifics on how ΔCDF and ΔLERF were calculated. The specific details are in "W.2.1 Methods Used to Determine Changes in Risk" with additional details provided in individual Fire Risk Evaluation documents.

a. The FRE assessments to support the VFDRs utilized the application FRANX. The quantification was based on a revision of the PAU condition to that of compliant conditions and determining the associated changes in CDF and LERF. The method of quantification is described in LAR Supplement Attachment W and PSA-WF3-03-01. No changes in modeling or methods were made to the peer reviewed version of the Fire PRA while being used in the FRE task. The process is documented, supporting this conclusion in each FRE calculation.

b. One new human action was modeled as a result of the Fire Risk Evaluations. The action is to locally trip the RCP motor breakers (this action applies to fires in RAB 1, RAB 7, RAB 8 and the Turbine Building). Instrument failure is not specifically addressed as the action is completed outside the control room at local breakers.

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. ML090410014), 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

Prior to the crediting of an Electric Raceway Fire Barrier System (ERFBS), the adequacy of the installation and fire endurance rating was verified. Only one ERFBS (in RAB 6) is credited in the FPRA model. In RAB 6 a 3M 'fire wrap' ERFBS protects power cables for Component Cooling Water Pump CCMPMP-0001B. The credited wrap is identified for continued monitoring as part of the fire protection program. 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 RAB 6. 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

Two modifications listed in Attachment S (of the LAR Supplement) are credited in the Fire PRA. A 3M fire wrap in RAB 6 is directly credited in the FPRA model. The wrapped cables in RAB 6 are assumed protected from fire in the base FPRA model. This wrap is physically installed and the adequacy of the installation and fire endurance rating has been verified. The second modification credited in the FPRA model involves the removal of secondary combustible material in RAB 27. This model assumes the secondary combustibles are no longer present in RAB 27.

In addition to the two listed modifications, an implementation item involving nitrogen accumulators supporting specific CCW and EFW valves is directly credited in the FPRA. No physical change to the accumulators is required in the implementation item. The implementation item reduces the allowable leakage of the system to ensure the accumulators can support valve operation through the 24 hour mission time. The fire wrap in RAB 6 (item S1-5), the removal of combustibles in RAB 27 (item S1-14), and the nitrogen accumulators allowed leakage (item S2-22) are in the Attachment S of the Waterford 3 LAR Supplement.

The delta risk values listed in Attachment W of the LAR Supplement do not credit any of the planned modifications listed in Attachment S with the exceptions noted above.

The fire PRA model will be updated within six months after the completion of all the modifications listed in the Attachment S. See Implementation Item S2-22 in Attachment S of the LAR Supplement.

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-

Both the Waterford 3 IEPRA and FPRA have been revised/updated since the last full scope peer reviews were completed (and since the original LAR submittal). The IEPRA model update was a periodic update to resolve open items (peer review findings) and update model data. No 'PRA Upgrades' (methodology changes) were part of the model revision; therefore no additional peer reviews were required. All methods used in the IEPRA have been peer reviewed. The LAR Supplement Attachment U provides details on IEPRA quality related to the most recent peer review findings.

The FPRA model has also been revised/updated since the last full scope peer review and since the original LAR submittal. The FPRA revision includes 'PRA upgrades'. Two separate focused scope peer reviews were conducted on the changes made to the FPRA analyses. The focused scope reviews were consistent with the guidance in ASME/ANS-RA-Sa-2009, as endorsed by Revision 2 of Regulatory Guide 1.200. The findings (and corresponding dispositions) of the focused scope peer reviews are included in LAR Supplement Attachment V.

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-W3-05-003, "Waterford 3 FPRA Quantification Model Preparation and Database Development Report," pages 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

The HRA methodology originally applied at Waterford 3 for the FPRA (for the original LAR submittal) has been revised. The revised approach is a simplified one that applies a fire adjustment factor to HFE values. A factor of 10 was chosen for the adjustment at the individual and combination level for actions where the probability of failure was less than 1.0E-2 and in particular combinations with 1.0E-5 or lower probabilities. No HFEs were screened out. All outside the control room actions except one (which was added during FRE process) were assumed failed due to fire.

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 internal events PRA (IEPRA) and Fire PRA both used these HRA methods (with the FPRA simply including a factor of 10 to the IEPRA value for each action). 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 IE PRA is described in detail in calculation PSA-W3-01-HR, "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 IE PRA. 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

The MCR Abandonment Analysis was revised and a 0.1 CCDP estimate is no longer applied. A detailed assessment of all actions needed to abandon the MCR was developed. The actions were then evaluated using the HRA Calculator approach discussed above.

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 3 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 3 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 3 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.

RAI PRA 15

Since cable damage thresholds and horizontal flame spread rates along cables are based on the type of cable (thermoplastic versus thermoset), please describe why a zone of influence (ZOI) is based on Institute of Electrical and Electronics Engineers (IEEE)-383 qualification rather than the type of cable employed as a decision step. An IEEE-383 qualified thermoplastic cable is still subject to the lower damage threshold (-205 degrees Celsius (°C)) and higher horizontal flame spread rate (-0.9 millimeters/second (mm/sec)) of thermoplastic cables. It further appears that IEEE-383 qualification rather than type of cable is being used to determine hot gas layer (HGL)-ZOI. A temperature of 329°C is cited as the damage threshold for IEEE-383 qualified cables, which would be appropriate only if they are thermoset, regardless of IEEE-383 *qualification. (See PRA-W3-05-005, "Waterford 3 FPRA Multi-Compartment Analysis and HGL ZOI Evaluation," pages 2-5 and 2-6, Figures 2-2, 69 kilowatt (kW) heat release (HR) Screening, and 2-3,702 kW HRR Screening; page 2-7, §2.6, Conclusions; page 3-2, §3.1, Purpose; pages 3-4 and 3-5, Figures 3-2, 69 kW HR Screening, and 3-3, 702 kW HRR Screening; PRA-W3-05-006, pages 2-8 and 2-9, §2.2, Assumption 8; pages 4-1 and 4-2, §4.0, Damage Criteria.) Please provide reanalysis that applies the correct assumptions.*

Also, in PRA-W3-05-006, page 10-1, §10.2, Cable Tray Propagation, the statement that thermoset materials do not propagate flames is incorrect. There is a horizontal spread rate of -0.3 mm/sec, and vertical spread is also likely, as per NUREG/CR-6850, including frequently asked question (FAQ) 08-0049 in Supplement 1, and the recent results from CHRISTIFIRE tests, NUREG/CR-7010, "Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE)" [draft for comment (ADAMS Accession No. ML 102700336), dated October 31, 2000; NPA version (ADAMS Accession No. ML 120540103) dated May 31, 2012).] Assuming no vertical propagation because Waterford has thermoset cables would be incorrect. Please provide reanalysis that applies the correct assumptions.

<u>Waterford 3 Response</u>

The design specifications for Waterford 3 cables required IEEE-383 qualification. The materials of construction of the cables are consistent with thermoset performance which was the basis for the determination for the FPRA. Findings from supplemental plant walkdowns following the audit documented in PRA-W3-05-022 support this conclusion based on a sampling of raceway cabling. The walkdown did confirm the existence of a very limited number of cables of a thermoplastic nature installed by a design change. These cables are associated with low voltage signals for multiplexer units and are located in non-safety related raceways in several physical analysis units (PAUs).

The majority of these cables are located in PAUs where fires are conservatively assumed to result in a complete loss of the PAU and the cable damage threshold is not a critical parameter due to the assumed damage. These PAUs are the turbine building, cable spreading room and RAB 27. The revised evaluation treated the identified cables in a manner consistent with the guidance in NUREG/CR-6850 assuming a thermoplastic material property. In the remaining PAUs, the cable population in any specific raceway is between 1 and 5 cables which is estimated to have a small impact on any existing analyses. In many cases, the thermoplastic cables are shielded by thermoset cables within the tray which would preclude significant heating to the point of damage.

With respect to vertical burning, the walkdown found that the thermoplastic cables were located primarily in horizontal runs or contained in raceways that were covered. Thus, the potential impact from vertical burning would be limited.

<u>RAI PRA 16</u>

In PRA-W3-05-006, pages 4-3 and 4-4, §4.2, Sensitive Electronics, the screening-level failure thresholds for sensitive electronics, 3 kiloWatts per square meter (kW/m^2) and 65°C, are at least twice as low as those for thermoplastic cables, let alone thermoset (330°C and 11 kW/m^2) which is the assumed base cable type. NUREG/CR-6850 allows a "relaxation" of the temperature threshold to 82°C if the sensitive electronics have been "qualified" at that temperature. Please provide a fire phenomenological/heat transfer calculation to confirm that "[e]:xtrapolating this argument to cabinets which are not 'adjacent,' it can easily be seen that separation distances beyond the ZOI associated with cable damage [presumably thermoset, which is less conservative than thermoplastic] are more than sufficient to preclude damage to sensitive electronics housed in cabinets." Also, please provide the basis for assuming that all sensitive electronics are contained within "sealed" cabinets, and therefore presumably immune to smoke *damage when the smoke originates from outside the cabinet. (See PRA-W3-05-006, page 4-5, §4-3, Smoke Damage.*)

Waterford 3 Response

Expanded ZOIs for sensitive electronics are explicitly accounted for in the revised analysis and are no longer screened. The fire scenario development for Waterford 3 FPRA was revised to include considerations for sensitive electronics. Revised calculations to determine fire scenario parameters: ZOIs, critical temperature, and time to damage, include such parameters for both normal fire damage and for solid-state ('sensitive') electronics (See PRA-W3-05-030, PRA-W3-05-006F).

RAI PRA 17

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 3 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 transient fire methodology originally applied at Waterford 3 for the FPRA (for the original LAR submittal) has been revised. The modeled heat release rates (HRR) for analyzed potential fires in the original submittal 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 limited the transient fire HRR for the baseline analysis. The transient fire scenarios were all revised to remove this UAM. All transient case HRRs were adjusted to a more conservative value (98^{th} percentile – 317 kW) consistent with NUREG/CR-6850.

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 was developed (expanded from the baseline ZOI) to meet the suggested criterion in NUREG/CR-6850 and reevaluated for the transient cases. As a result of these changes, all transient scenarios were reevaluated and new results generated. Additionally, no credit is taken for CPTs in the updated analysis as part of the updated LAR Supplement.

Fire modeling treatments used in the update analysis include the use of NUREG-1805 Fire Dynamics Tools or FDTs spreadsheets for the primary analysis of fixed and transient ignition sources for ZOI and HGL impacts. The revised fire scenarios reports (PRA-W3-05-006F- fixed ignition sources and PRA-W3-05-006T – transient ignition sources) include the details on how fire spread is assessed in the updated model.

Other more detailed fire modeling was performed for the main control room abandonment study using a Consolidated Model of Fire Growth and Smoke Transport (CFAST) model of the main control room for various fire scenarios as contained in the original LAR submittal and again used as the fire modeling basis in the LAR Supplement for the main control room abandonment timings. The revised multi-compartment analysis in the LAR Supplement (PRA-W3-05-005) contains fire modeling via the FDTs as well as a CFAST model of PAU RAB 7 for various fire scenarios.

RAI PRA 18

In the Transition Report, page V-14, Attachment V, Table V-1, Fire PRA Peer Review F&Os, the disposition for F&O PP-A 1-01 states that "Transient sources would not include sufficient loading to challenge the 2-hour rating." Please describe whether this considers the possibility of transient HRRs more severe than those assumed in the Unreviewed Analysis Method that used the 69 kW HRR for many transient combustible fires. Please describe whether the Sensitivity Analysis of the method addressed this possibility. Specifically, in PRA-W3-05-009, "Waterford 3 FPRA Sensitivity Evaluation to Address Alternate Methodologies Report," at least some of the combustible control limits for the Levels 1 and 2 transient fires seem more than sufficient to produce a higher HRR than 69 kW, possibly as high as the 317 kW, 98th percentile value recommended in NUREG/CR-6850. Please describe whether at least one of the following transient combustibles administratively controlled in Level 2 Areas could produce a higher HRR than 69 kW:

a. *100 pounds (lbs) of fire retardant lumber (contribution to fire limited based on fire retardant characteristics),*

b. *25 lbs of loose ordinary combustible or plastics,*

c. *5 gallons (gal) of combustible liquids in approved containers (contribution to fire will be limited based on use of approved container),*

d. *1 pint of flammable liquid in approved containers (contribution to fire will be limited based on use of approved container), or*

e. *one 20-ounce (oz) flammable aerosol can (material within can would require fire due to fixed ignition source or other transient to become flammable, typical ignition source associated with overloaded temporary cable or sparks/slag from welding are unlikely to involve flammable material within an aerosol can).*

Please describe the basis for a priori dismissal of all the above combustibles.

Further, in the Transition Report, on pages V-17 and V-18, §V.2.1, Reduced Heat Release Rate for Transient Combustible Fires; unless there is a physical constraint on the amount of combustibles (i.e., not just "administrative controls"), the Sensitivity Analysis should consider the possibility of higher HRRs, such as 317 kW, in Levels 1 and 2 areas. Not only total CDF and LERF, but also ΔCDF and ΔLERF need to be addressed, both plant-wide (cumulative) and per PAU (or other unique subdivision). On pages V-18 and V-19, §V.2.2, Adjustment Factor for the Transient Combustible Fire Ignition Frequency. the citing of FAQ 08-0044, related to mechanical feedwater pump (MFWP) oil fires, should be FAQ 08-0048.

Related to these, PRA-W3-05-009, "Waterford 3 FPRA Sensitivity Evaluation to Address Alternate Methodologies Report," contains extensive discussion of the use of a 98th percentile HRR of 69 kW for screening and modeling of transient combustible fires in selected locations, specifically the following: (Note that in Appendix E, multi-compartment analysis (MCA) Initial Screening, of PRA-W3-05-005, "Waterford 3 FPRA Multi-Compartment Analysis and HGL ZOI Evaluation Report," numerous fire zones listing "transient" as an ignition source type use the 69 kW value as the HRR for screening evaluation.)

i. *On pages 2-7,15 of the 21 tests used in NUREG/CR-6850 to generate the transient combustible fire HRR curve are dismissed as inapplicable or inappropriate. Fire testing in a relatively small enclosure (at least compared to nuclear power plant (NPP) rooms) is common practice to measure HRR (oxygen consumption calorimetry). Dismissal of these room tests leaves only six on which to base the HRR profile. For a fire up to -300 kW, the room will have no significant effect on fire behavior. A HGL is needed to approach the flashover conditions where the feedback effect begins to have an impact. Further, tests should not be discarded because the fires occurred against a wall or corner. The effect of the wall or corner on the actual burn behavior is relatively small and more complicated than "intensifying the fire." Proximity to walls may actually limit fire intensity during some stages by restricting the free-flow of fresh air (hence oxygen) to the fire, and a cold wall can absorb a lot of heat during the first few minutes. Any feedback is not a factor for some time. The use of an "accelerant" to ignite a test fire does not justify dismissing the test outright, as the HRR contribution from the accelerant would be minimal once the fire develops. The basis for discarding fire tests due to "plastic pool fires" is also questionable since several of the tests included a plastic bucket or trash can that melted down into a pool. Also, dismissal of the test fires ignited with a liquid, generally in small quantities that burn off quickly, ignores that use of a liquid as igniter is common for fire testing.*

ii. *On pages 10-11, a review of the impact on CCDP based on fixed ignition sources in several of the Level 3 and 4 locations is used at least as partial justification for not postulating any "worse" potential CCDP locations ("pinch points") for transient combustible fires. Describe if the targets potentially impacted only by the fixed ignition sources, bound the highest CCDP possible in the area, (Le., describe if it is possible for a transient combustible fire to be particularly located where it could damage a pinch point with a CCDP higher than any for the fixed ignition sources) Also, even if a transient fire could be located where it impacts a target already modeled for impact by a fixed ignition source, describe if the additional fire frequency from the transient fire impacting that target would merit inclusion in the risk estimate. If it was previously screened out due to the lower assumed HRR, its contribution would not be included in the existing estimate with the fixed ignition source, so it would have to be added as part of the Sensitivity Evaluation. In addition, the statement regarding the turbine generator building (TGB) that "increasing the size of this fire will not alter the CCDP or CDF for this scenario" may only be partially true. While the CCDP would not increase, being already set at the maximum for the base scenario, the CDF could increase if there is additional* *contribution via the frequency due to potential damage from a transient fire with a greater HRR, if that fire was previously dismissed when screened out due to lower HRR.*

iii. *It is stated on pages 19-20 that "Routing of the cables of concern away from these scenarios is at an elevation of greater than 12.4', ensuring that a 317 kW fire will not be able to damage the cables." Describe the ZOI for the 317 kW fire. Describe if there any other cables or combustibles within the ZOI that could be ignited and, through fire propagation, damage the cables of concern if not suppressed. In addition, when estimating the reduced CCDP =3.96E-4 for fires in the "other" locations, describe whether all five listed components are excluded simultaneously (Electrical Switchgear (ESWGR)-31A and RC-MPMP-0001A through 2B)? Some of the nine scenarios affect as few as only two of these five components, such that at least three others could be affected and should be included in the CCDP. If the CCDP = 3.96E-4 represents exclusion of all five, it is only appropriate if all five components are affected by a scenario. Those scenarios affecting less than all five would be expected to have a CCDP between 3.96E-4 and the maximum of 0.106. Also, the assumption of an automatic reduction in CCDP by a factor of 2 appears arbitrary, especially in light of the potential underestimate of CCDP for some of the "other" scenarios.*

iv. *On pages 28-34, §2.5.2, Transient Combustible HRR and Transient Adjustment, citing the zones in RABXX as immune to the use of the 317 kW HRR analysis is questionable given the previous comments on pages 2-7.*

<u>Waterford 3 Response</u>

This request (and corresponding sub-requests) is based on an unapproved method (UAM). The method in question has been removed from the revised analysis and the RAI request and previous response are no longer applicable based on the revised FPRA model, methods and documentation.

The baseline analysis (for the original LAR submittal) used the 69 kW HRR for transient combustible fires and did not consider the possibility of a more severe HRR. All transient fire scenarios have been revised using more conservative (and NUREG/CR-6850 compliant) HRRs. This UAM has been removed from the WF3 FPRA analysis and results.

A review of the administrative control limits identified that the 69kW HRR is not bounding. The re-evaluation considered the aforementioned combustible listing when re-evaluating the transient fire scenarios completed with RAI PRA 02 response using 317 kW, the 98th percentile HRR value recommended in NUREG/CR-6850, for transient fires. This 317 kW HRR was also used for any transient combustible scenario within the administratively controlled Level 2 areas and it is expected that the 317 kW transient fire, HRR would encompasses transient combustible loadings within these areas. Furthermore, the basis used for excluding the higher overall HRRs than 69 kW in the previous Waterford 3 Fire PRA model was not used during the completion of the transient fire scenario re-evaluations developed to support the LAR Supplement. It would be expected for overall HRRs above 69 kW to be present for items such as the list of combustible/flammable materials in the above RAI text, and that the use of the NUREG/CR-6850 overall transient fire source HRR of 317 kW better encompasses potential transient combustible materials. The new analysis is based on the guidance in NUREG/CR-6850 and does not include any reduction factors.

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 D-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 PRA-W3-05-007. The peer review conducted for the Waterford 3 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. Though no quantitative uncertainty was developed, a sensitivity assessment of the selection of the source frequencies (NUREG/CR-6850 Supplement 1 vs NUREG/CR-6850) was completed.

RAI PRA 20

Please describe the basis for the selection of 69 kW and 702 kW as the HRR groups for the HGL screening and whether the HRR increased for the potential contribution due to fire spread to other combustibles, such as propagation horizontally or vertically along cables. Also, since sensitive electronics are assumed to be potentially damaged at 82°C (See Appendix S of NUREG/CR-6850), please describe whether attainment of the 80°C threshold was considered for failure for sensitive electronics (vs. the next screening level of 220°C). (See PRA-W3-05-003, pages 2-1 and 2-2, §2-2, Methodology, and §2.3, Background/Analysis Inputs.)

Waterford 3 Response

The request is based on analyses that have since been revised. The choice of HRRs and sensitive electronic damage threshold were based on the application of UAMs in the analysis that supported the original Waterford 3 NFPA-805 LAR submittal. These methods are no longer included in the updated Fire PRA. The revised fire scenarios include higher HRRs (compliant with NUREG/CR-6850 methodology) and expanded ZOIs for sensitive electronics.

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

The analysis that supported the original Waterford 3 NFPA-805 LAR submittal did not consider secondary ignition or fire spread from secondary ignition. All fire scenarios have since been revised and secondary ignition was evaluated for all fixed sources and transients. The expansion of modeled fire impacts to targets beyond the nominal ZOI but within range of an expanded fire is included for several scenarios. The updated documentation (PRA-W3-013, PRA-W3-05-006F, and PRA-W3-05-006T) describes how fire spread is assessed in the updated model.

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-6850) 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-W3-05-006, 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

This RAI and previously submitted response are no longer applicable. The updated FPRA analysis utilized NUREG/CR-6850 Appendix L methodology.

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 Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The updated analysis includes detailed fire modeling (as well as the application of higher HRRs for transients). The RAI request and previous response are no longer applicable based on the updated FPRA model and analysis methods.

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]/[1.5E-5]), including fragilities, such that CDF ~ (1.5E-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 3.

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 3 in GI-199 is so conservative, a slightly more realistic seismic CDF is calculated using the Waterford 3 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 3 is 9.02E-7/yr. Since LERF is taken to be an order of magnitude less than CDF, the Waterford 3 seismic LERF is 9E-8/yr.

RAI PRA 25

In the Transition Report, page V-19, §V.2.3, Adjustment Factor for the Hot-work fire Ignition Frequency, credit for administrative controls during hot work is embedded in the NUREG/CR-6850 hot-work fire frequencies. This Unreviewed Analysis Method was not accepted by the consensus industry-NRC panel and was replaced with an alternative approach. As such, it should not be retained in the FPRA. "A sensitivity analysis was performed by increasing the hot work frequency reduction factor for 'cable fires caused by welding and cutting' and 'transient fires caused by welding and cutting' from 0.01 to 1.0. The results of this evaluation show that the CDF increased from 3.42E-5/yr to 4.24E-5/yr. This represents an overall CDF increase of approximately 24 percent. This evaluation shows the results are still within the guidelines for transition found in NRC Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011 (ADAMS Accession No. Ml100910006), and RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, December 2009 (ADAMS Accession No. Ml092730314). Provided this was done for all hot work PAUs, this sensitivity is acceptable. However, LERF, ΔCDF, and ΔLERF sensitivity results should also be provided. Related concerns are as follows:

a. *In W3F1-2012-0005, page 3 of 7, SQ 3, Alternate Method #3, Adjustment Factor for Hot Work Ignition Frequency, assuming this refers to Item S1-5 in the LAR, please describe to what proposed procedural modifications this reference applies. S1-5 cites a proposed installation of 3M electrical raceway fire barrier system (ERFBS) in reactor auxiliary* *building (RAB) 5 and 6 only. Also, please describe whether the reference to Item S2-8 is for procedure EN-DC-127 and whether hot work will now be prohibited versus limited/controlled, in RAB 5 and If not, then please describe whether the 0.01 reduction factor is still being applied and if yes, please provide the Sensitivity Results for these locales without this credit.*

b. *b. In PRA-W3-05-006, page 8-2, §8.1, Suppression and Detection Random Failure Probabilities; page 8-3, §8.2, Hot-work fire Scenarios, a failure probability of suppression during hot work procedures of 0.01 is cited, which is apparently based on the original Unreviewed Analysis Method. Please provide the methods Sensitivity Analysis that uses the modified, accepted version of this method, as per the consensus industry-NRC panel.*

c. *In PRA-W3-05-009, pages 24-26, §2.4, Adjustment Factor for the Hot Work Ignition Frequency, the hot work correction factor of 0.01 was applied only to scenarios in RAB 5 and RAB 6, as listed in Table 2.4.1, and the increases in ignition frequency and CDF are slightly less than a factor of 10 when the correction factor is increased to 1 (a 100-fold increase). Please describe whether it is correct to assume that the transient fire frequencies listed in Table 2.4.1 include contributions from non-hot work transient fires, thereby accounting for the overall increase by "only" 24 percent.*

Waterford 3 Response

The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The updated analysis no longer relies on Unreviewed Analysis Methods (UAM). The hot work adjustment factor is no longer applied to fire scenarios. This RAI request and previous response are no longer applicable based on the updated FPRA model and analysis methods.

RAI PRA 26

In the Transition Report, pages V-19 and V-20, §V.2.4, Adjustment Factor for Electrical Cabinet Ignition Frequency, although apparently limited in use only to screening, this Unreviewed Analysis Method is not accepted by the NRC. The screening should be re-performed with no credit for this factor, or using fire modeling phenomenology that is generically bounding. Also, in W3F1-2012-0005, page 3 of 7, SQ 3, Alternate Method #4, Adjustment Factor for Electrical Cabinet Ignition Frequency, it is stated that "... [T]his method has only negligible impact on the CDF and LERF. Therefore, there is no change in CDF or LERF being reported for this method." The LAR assumed the adjustment factors to be doubled, vs. increasing by factors of 10 and 100, respectively, to correspond to eliminating the credit altogether. Please describe whether this conclusion applies to the latter case.

Related concerns are as follows:

a. *In PRA-W3-05-005, page 1-6, Appendix I, Summary of Bin 15 Conditional Probabilities, the use of conditional probabilities of 0.1 and 0.01 for SWGR, load centers, motor control centers (MCCs), is a deviation from NUREG/CR-6850 that requires justification or reanalysis via a Methods Sensitivity Evaluation where it is not credited.*

b. *In PRA-W3-05-009, page 27, §2.5.1, Adjustment Factor for Electrical Cabinet Ignition Frequency, the sensitivity analysis, which increases the adjustment factor only two-fold, needs re-evaluation via a methods Sensitivity Evaluation removing this credit entirely, or one based on a fire phenomenological approach consistent with NUREG/CR-6850 that establishes a bounding value that may be <1 for the adjustment factor.*

Waterford 3 Response

The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The updated analysis no longer relies on Unreviewed Analysis Methods (UAM). The electric panel adjustment factor is no longer applied to fire scenarios. This RAI request and previous response are no longer applicable based on the updated FPRA model and analysis methods.

RAI PRA 27

In the Transition Report, pages V-20 and V-21 , §V.2.5, Severity Factor for Pump Oil Fires, the consensus industry-NRC panel did not accept this split in favor of one using 90 percentl10 percent. Subsequently, the NRC questioned the propriety of even this split as well as the estimate of the fraction of applicable spills. Supplement this sensitivity analysis with the 90 percentl10 percent split with one accounting for any additional NRC clarifications related to fire events data to account for in the analysis. Please provide CDF and LERF and ΔCDF and ΔLERF. (See also the entry for Oil Spill in Table 7-1 in PRA-W3-05-006, page 7-2, §7.0, Severity Factor.)

Related concerns are as follows:

a. *In W3F1-2012-0005, page 4 of 7, SQ 3, Alternate Method #5: Severe Pump Oil Fire Effects, Item S1-5 in Att. S, Table S-1, Plant Modifications, proposes to install 3M ERFBS in locale RAB 2, et al. Please describe if this was not credited before in the LAR when reporting the risk and delta-risk for the RAB2 scenarios (total = -1.41E-9). Please re-evaluate the results for RAB2 crediting Item S1-5 in light of the NRC clarification of the panel resolution.*
b. *In PRA-W3-05-009, pages 35-40, §2.6, Split Fraction for Pump Oil Severe Fire, the sensitivity analysis compares the Unreviewed Analysis Method split fractions against those from FAQ 08-044, which applies solely to MFWPs. The appropriate Method Sensitivity would compare against the NUREG/CR-6850 split fractions, where the contribution from large and very large fires (0.0196 and 4.00E-4, respectively) are combined into one (very large fires = 0.020). The result would be a change where the severity factor for moderate fires would drop to zero while that for severe fires would rise from 4.0E-4 to 0.020. The effect would appear to be bounded by an increase for the severe fire CDFs by (0.0200.00040)/0.00040 = 49. This would be somewhat offset due to the moving of the moderate fire, with its 0.0196 split fraction and typically lower CCDP (than the severe fire), into the severe fire group. For example, in Table 2.6.2, dominant Scenarios E017, E017M and E017S in RAB2 combined for an original CDF = 8.01E-7. With the appropriate method Sensitivity, the new CDF would be (1.34E-4)(0.506)(0.020) = 1.36E-6, -17 times higher (also -53 percent higher by itself than the total CDF calculated in Table 2.6.2 for all scenarios, 9.04E-7). (See also pages 41-43, §2.7, Split Fraction for DG Room Fire Evaluation.)*

Waterford 3 Response

The application of split fractions for pump oil fires was revised in the Waterford 3 FPRA. The updated analysis utilized the suggested 90/10 split. Due to the scope of changes to the model, the results based on this change alone are not presented. No sensitivity is required as the updated analysis utilizes the suggested methodology.

Related Concern (a):

The 3M ERFBS fire wrap barrier modification is not a result of fire risk evaluations, but is credited in the fire PRA model for RAB 6 only. The results of the fire risk evaluations did not identify a need to credit 3M ERFBS in RAB 2 in order to meet the acceptance criterion.

Related Concern (b):

The updated analysis addresses this concern by utilizing the 90 percent / 10 percent split methodology.

<u>RAI PRA 28</u>

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 1 E-6 become potentially risk significant if set to a minimum of 1 E-5.

<u>Waterford 3 Response</u>

The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The updated analysis no longer uses a floor for joint HEP values. A simpler method of applying a factor of 10 to human actions for fire is used. A factor of 10 was chosen for the adjustment at the individual and combination level for actions where the probability of failure was less than 1.0E-2 and in particular combinations with 1.0E-5 or lower probabilities. This RAI request and previous response are no longer applicable based on the updated FPRA model and analysis methods.

<u>RAI PRA 29</u>

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.

<u>Waterford 3 Response</u>

The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The updated analysis is compliant with NUREG/CR-6850 and no longer relies on Unreviewed Analysis Methods (UAMs). This RAI request and previous response are no longer applicable based on the updated FPRA model and analysis methods supporting the LAR Supplement.

<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 Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The generic factor applied in the prior version was removed. The revised analysis utilizes the factors provided in NUREG/CR 6850, Supplement 1 and the value is selected in accordance with the fire source. This RAI request and previous response are no longer applicable based on the updated FPRA model and analysis methods supporting the LAR Supplement.

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 document listed in the request has been updated and split into two documents (one for fixed and one for transient ignitions sources). Neither of the updated documents include a table citing industry standards or FAQs. The treatment of Hot Shorts in the revised analysis is within the bounds of FAQ 08-0051 and is consistent with NUREG/CR-6850 guidance. There are no cases where hot short durations were applied (no recovery of hot shorts is credited)

RAI PRA 32

PRA-W3-05-006, page 2-8, §2.2, Assumptions, provides the basis for Assumption 5. For a fire started in a single cable bundle, please discuss whether or not the resulting fires would involve multiple cables in the panel. If so, please describe whether the HRR and other fire effects would be more severe than would be expected for Assumption 5, thereby leading to shorter abandonment times (and higher risk). (See also, page 13-2, §13.1, MCR Analysis; pages 13-4 and 13-5, §13.2.1, MCR Abandonment Times; page 13-21, §13.2.2, MCRAB CCDP Determination)

Waterford 3 Response

The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The updated analysis no longer relies on 'Assumption 5' that is referenced in this request. The MCR Analysis (PRA-W3-05-028) and Fire Scenarios Report (PRA-W3-05-006F) have been revised, differentiating between single and multiple cable bundle fires with different abandonment times for control room abandonment scenarios (PRA-W3-05-026). This RAI request and previous response are no longer applicable based on the updated FPRA model and analysis methods used to support the LAR Supplement.

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.*

<u>Waterford 3 Response</u>

The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement including the modeling of transient fires.

Guidance is given in PRA-W3-05-006F for treatment of "sealed" cabinets based on information contained in NUREG/CR-6850 which indicates the potential for propagation of electrical panel fires for cabinets that are not vented will not propagate a fire. However, all cabinets are treated as having the ability to propagate a fire based on their resultant zone of influence impacts in the WF3 fire PRA. For analysis of cabinet fires, the resultant fixed source heat release rate zone of influence is used on a case-by-case basis to determine if any potential secondary combustibles are within the resultant zone of influence for the particular ignition source and it combustibles are present, ignition of the secondary combustible(s) is based on the zone of influence criteria as listed in PRA-W3-05-006F. All fire scenarios have since been revised and secondary ignition was evaluated for all fixed sources and transients. The expansion of modeled fire impacts to targets beyond the nominal ZOI but within range of an expanded fire is included for several scenarios. The revised fire scenarios reports (PRA-W3-05-006F - fixed sources and PRA-W3-05-006T – transients) include the details on how fire spread is assessed in the updated model.

The updated analysis considers that transient fires are possible over all of the available floor space. Also, the updated analysis uses the NUREG/CR-6850 guidance for using the 98th percentile 317 kW heat release rate for transient fire sources versus the 69 kW fire credited previously. With respect to elevation of transient fires, Assumption 11 from the previous methodology is no longer employed. The zone of influence for a transient fire source of 317 kW is developed in PRA-W3-05-013, and that zone of influence is applied for the particular scenario of interest by an analyst using judgment as to the elevation of the transient fire source (i.e. floor level or some other elevation) on a case-by-case basis in PRA-W3-05-006T. In general, the available floor space transient scenarios are based on floor level as no substantial amounts of transient combustibles were noted during the numerous walkdowns as existing at higher elevations and the floor elevation is the expected location for temporary cleaning or maintenance items brought into a particular PAU. With regard to trash can fires, it was observed during the various walkdowns that trash cans contained limited combustible materials and thus the subsequent fire scenario would be expected to ignite and occur at a low elevation near the floor level, no instances of a trash can fire requiring elevation above floor level were observed for the limited amount of trash cans present in the various PAUs.

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.

<u>Waterford 3 Response</u>

The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The updated analysis of oil fires (PRA-W3-05-029 & PRA-W3-05-006F) considers structural collapse from large oil fires based on methods described in FAQ 08-0044 for very large oil fires. An assumed CCDP value of 0.01 is no longer applied, using a calculated CCDP instead.

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

The report cited in the above request has been revised and superseded. There are now two fire scenario reports (PRA-W3-05-006T- transient and PRA-W3-05-006F – fixed sources) that document the heat releases rates used for all scenarios. In the updated scenarios, no distinction is made between heat release rates of vented and unvented cabinets.

RAI PRA 36

In PRA-W3-05-006, pages 13-4 and 13-6 through 13-20, § 13.2.1, MCR Abandonment Times, both NUREG/CR-6850, Appendix P, and FAQ 08-0050 in Supplement 1 limit the minimum non-suppression probability to 0.001. Zero is not permitted by either reference. The use of zero leads to underestimation of the summed bin frequencies in Tables 13-1 through 13-15 (e.g., an additional [1 -0.008] [0.001] =9.92E-4 would accrue in Table 13-2). Please provide a reanalysis using 0.001 as the minimal NSP. Also, the correspondence between Tables 13-1 through 13-15 and Tables 6-1 through 6-6 in PRA-W3-05-006B, "Evaluation of the Unit 3 Control Room Abandonment Times at the Waterford Nuclear Station," appears to be misaligned in some cases (e.g., 13-2, 13-6, and 13-7 go with 6-2, which is Case 2, not Case 1; 13-8,13-9, 13-10, and 13-11 go with 6-4 (Case 4) or 6-5 (Case 5), not Case 1; 13-13 and 13-14 go with 6-5, which is Case 5, not Case 1; 13-15 goes with 6-5 (Case 5) or 6-6 (Case 6), not Case 1). Please correct the discrepancies.

Waterford 3 Response

The use of a zero non-suppression probability has been removed and the limiting value of 0.001 applied as the minimum value in the revised analysis. The table values and misalignment noted in the RAI request have been superseded by the revised analysis.

RAI PRA 37

In PRA-W3-05-006, Appendix F, Beyond 6850 -Supplemental Factor Development, three of the Supplemental Factors presented (Hotwork Alignment, General Transient Alignment and Pump Fires Severity) have been processed through the Unreviewed Analysis Methods consensus industry-NRC panel, with modifications approved to the original proposals provided here. For each of these methods, use of an accepted method needs to be performed and the results provided (noting clarifications provided by the NRC on the latter two). The fourth, EDG Aggressive Fire Factor, has not previously been presented to the consensus panel as an Unreviewed Analysis Method, so remains unresolved as to its validity and proper use at this time. For this method, a Sensitivity Analysis using an accepted method needs to be performed and the results provided.

Waterford 3 Response

The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The updated analysis no longer relies on Unreviewed Analysis Methods (UAMs). The UAM in question was removed from the analysis as a part of the model update. The revised methodology no longer differs from NUREG/CR-6850.

The EDG aggressive fire split fractions has also been removed from the analysis. The updated analysis assumed that any EDG fire will result in a complete room burn unless the suppression system functions. The severity factor for the non-aggressive scenario is set to zero indicating that no smaller fires are allowed. The potential for suppression remains unchanged.

This RAI request and previous response are no longer applicable based on the updated FPRA model, documentation, and analysis methods supporting the LAR Supplement.

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

The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The updated discussion for Task 6 uncertainty (PRA-W3-05-007 Revision 2) does not describe ignition frequency and NSP values as containing significant conservatism. The uncertainty assessment for each Task is provided in section 4.2 of PRA-W3-05-007.

With regard to CPTs (Current Potential Transformers), no credit is taken in the circuit failure analysis.

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

The Waterford 3 Fire PRA was substantially revised in response to RAIs. The update included conversion from the FRANC software to FRANX.

In the FPRA quantification (using FRANX) 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; standard importance measures show the importance of the non-fire-affected components only. Due to these known limitations in the FPRA quantification method, standard PRA importance measures were intentionally omitted from the updated Summary Report (PRA-W3-05-007).

RAI PRA 40

In PRA-W3-05-009, pages 44-45, §2.8, Evaluation of Combined Impact, the evaluation results, especially for §2.1 and §2.4 through §2.6, and the resultant combination for all, need to be re-evaluated. Also, an evaluation of the effect from these Sensitivity Analyses on total CDF and ΔCDF and fire LERF and ΔLERF for their plant-wide combination is warranted. The extent to which screening phases of the analyses would be affected by the various sensitivity calculations needs to be addressed. Also, the potential dependence among sensitivity parameters may make it necessary to evaluate some factors together, rather than individually.

Waterford 3 Response

The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The updated model results for total CDF, LERF, ΔCDF and ΔLERF are contained in Attachment W of the LAR Supplement. The results presented in Attachment W include the aggregate effect of all the changes incorporated during the FPRA revision (including but not limited to removal of all UAMs from the analysis).

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 7A-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 7D, 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

The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. A part of this update included revisiting the identification of VFDRs. The changes to the VFDR list were the result of a more thorough understanding of the methodology. The updated documentation on VFDRs is more thorough and describes how each is treated in the FPRA model. Table B-3 in the LAR Supplement is clearer than the one in the original LAR submittal. A supporting document was also developed to assess each VFDR and detail how it is treated in the Fire PRA model (including detailed justifications on VFDRs that are not modeled). This document is PSA-WF3-03-01, "Waterford 3 - Methodology for Addressing VFDRs in the Fire PRA and NFPA-805".

LAR Attachment W has been revised and resubmitted with the Waterford 3 NFPA 805 LAR Supplement. The updated Attachment W no longer uses epsilon for very low delta risk values. All areas with delta risk evaluations have a numerical value (0.00 for ones that had zero delta risk at truncation level). All areas that did not have FRE calculations developed (deterministic areas) have “N/A” in the delta risk column in Attachment W Table W-2.

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 SSD-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 SSD-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. LAR Attachment S has been revised and resubmitted with the LAR Supplement. Based on the contents of the updated Attachment S this RAI request and previous response are no longer applicable. The modifications that were the subject of this request were changed based on the revised FPRA. The modification item S-1 is ranked 'Low' and none of the listed items are in the PRA model. Item S-2 has been deleted.

b. LAR Attachment S has been revised and resubmitted with the LAR Supplement. The previous S1-3 Modification is no longer necessary due to the results of the updated Fire PRA. Based on the contents of the updated Attachment S this RAI request and previous response are no longer applicable.

c. LAR Attachment S has been revised and resubmitted with the LAR Supplement. The previous S1-4 Modification is no longer necessary due to the results of the updated Fire PRA. Based on the contents of the updated Attachment S this RAI request and previous response are no longer applicable.

d. LAR Attachment S has been revised and resubmitted with the LAR Supplement. The modification item S1-5 is ranked 'low' in the updated Attachment S. Plant fire protection and PRA personnel implemented the modification based on 'engineering judgment' early in the transition process. The Waterford 3 staff recognized the need based on known separation issues. The modification (3M fire wrap) is credited in the FPRA in only RAB6. 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 too 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 AB 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-B3-02, 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 B have a 1-hour lifetime for non-SBO 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 3 Accident Sequence (AS) analysis.
The ‘other phenomena’ referred to in the response refers to all phenomena in WCAP-16679-P that are relevant to the Waterford 3 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

b. Based on the revised internal events model this RAI and previous response are no longer applicable. The Waterford 3 Internal Events PRA model was updated following the original NFPA 805 LAR submittal. The update includes surveillance data (via operator logs) and no longer uses the PI system. The current data effort meets the standard for the listed SR requirements. Dispositions for each finding are documented in the Waterford 3 LAR Supplement Attachment U.

c. Based on the revised internal events model this RAI and previous response are no longer applicable. The Waterford 3 Internal Events PRA model was updated following the original NFPA 805 LAR submittal. Alternate alignments were added to the model to address the F&O. Plant-specific operational records were used to determine the time that components were configured in standby status. The model was revised to use conditional probabilities, as appropriate, for systems that have both running and standby equipment associated with them for various alignments.

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 3 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. Based on the revised internal events model this RAI and previous response are no longer applicable. The Waterford 3 Internal Events PRA model was updated following the original NFPA 805 LAR submittal. Non-proceduralized actions are no longer credited in the WF3 PRA and this element of the standard is fully satisfied.

f. Based on the revised internal events model this RAI and previous response are no longer applicable. Ultimate Heat Sink (including WCT and DCT) success criteria was developed and documented in the updated Internal Events PRA.

g. The battery rooms at Waterford 3 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. Credit for battery load stripping was intentionally not applied in the Internal Events PRA due to risk results being acceptable without such benefit. The Fire PRA is consistent with this modeling choice. The significance of not modeling load stripping for this LAR is related to fire-induced station blackout and loss of AC power (loss of offsite power is not a credited initiator for the fire PRA). By crediting load stripping for these scenarios, additional time would be available to restore offsite power which would increase the recovery probability thereby reducing CDF. However, recovery of off-site power is not credited in the Waterford Fire PRA model; therefore nullifying any such CDF reduction.

i. The Waterford 3 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. Based on the revised internal events model this RAI request and previous response are no longer entirely applicable. The updated internal events model includes a value for battery unavailability.

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).

For both the internal events and FPRA model updates flow diversion pathways were reviewed to determine if additional pathways needed to be included to address potential spurious actuation opening of power operated valves. As part of this review, the flow diversion pathways excluded due to the 1/3 rule were reviewed to verify that no pressure differential is present and that sufficient margin is built into the system flow. Considerations for extended time (up to 24 hours) for systems that meet the 1/3 criteria resulted in additions to the model. Flow diversion of the CCW and CCW Makeup systems could cause system failure. These failures were added to the model (both internal events and FPRA).

l. The MCR HVAC has been added to the internal events model during the model update. The disposition of this finding (SY-B13) was revised in the Waterford 3 LAR Supplement Attachment U. The quoted text is no longer included in the disposition. The impact of fires and HVAC function on control room abandonment is addressed in scenario development (PRA-W3-05-028).

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.

RAI PRA 45

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-W3-05-006, 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.*

Waterford 3 Response

a. The impact of loss of DC power on the ability to trip the RCP breakers is addressed in the WSES FPRA. A separate calculation to support the FPRA (PRA-W3-05-034 – "Revised WF3 Reactor Coolant Pump Trip Failure Mode Model for FPRA Including Trip Cables") was completed to evaluate the impact of fires on RCP related cables (including DC power cables).

b. The Component and Cable Selection Report (PRA-W3-05-002 Rev 1) discusses the method of identifying additional components from the internal events PRA (Sections 2.0 Equipment Selection, 2.1 IE PRA model review, 2.2 SSEL, and 2.4 MSO review). The Component Selection document (and Attachments) represents a systematic evaluation to ensure the FPRA model includes all high risk important components from the internal events model (including some not in the IE model). 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.

c. The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The updated model does not credit any self healing. This RAI request and previous response are no longer applicable based on the updated FPRA model, documentation, and analysis methods.

d. Both the FPRA and internal events PRA have been updated. Convergence evaluations were completed as part of each update. The truncation values used in the FPRA quantifications are 1E-06 and 1E-08 for CDF (CCDP) and LERF (CLERP) quantification, respectively. The table below shows the results of the test for convergence. As shown in the table below, the selected truncation and convergence meet the criteria of standard requirement FQ-B1. (Results below from PRA-W3-05-050)

Truncation	CDF	% Chg	LERF	%Chg
1.00E-04	1.40E-05		NA	NA
1.00E-05	1.49E-05	6.4%	NA	NA
1.00E-06 (baseline)	1.54E-05	3.4%	NA	NA
1.00E-07	1.60E-05	3.8%	NA	NA
1.00E-08 (LERF baseline)	1.62E-05	1.2%	2.87E-06	0.8%
1.00E-09	NA	NA	2.88E-06	0.3%

The FPRA convergence is at a much higher truncation limit than the IEPRA. Convergence occurs in the FPRA at 1E-6 which is the baseline quantification limit (for calculating CCDP values). The base CDF in the IEPRA is quantified at 1E-11 with convergence also demonstrated at 1E-11. The difference in convergence limits is partially due to the fire results being an order of magnitude higher than the base CDF results as well as the methodology differences. The IEPRA results are from initiating events (generally in the 1E-3 to 1E-5 range) with numerous random failures required to reach core damage. Fire results have fire ignition frequencies, which are generally higher than IEPRA initiating events, and numerous fire damage events accounting for additional failures. The number of random failures needed to get to core damage is much lower (including many scenarios with no random failures needed – 1.0 CCDP). The fewer random failures to get to fire related core damage is the primary reason for the difference in truncation/convergence limit disparity.

e. The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The revision includes quantitative uncertainty evaluation for both CDF and LERF results (as well as a documented reasonableness review of all results). This RAI request and previous response are no longer applicable based on the updated FPRA model, documentation, and analysis methods.

f. Plant specific fire suppression system unavailability data was developed as part of the FPRA update. Numerous system failures (outlier behavior) were identified that required the need to use plant specific values. FPRA supporting calculation PRA-W3-05-031 "Development of Fire Nonsuppression Factors for WF3 Fire PRA Scenarios" documents the development of the plant specific values. This RAI request and previous response are no longer applicable based on the updated FPRA model, documentation, and analysis methods

g. RHFPUMPOFP does not have a detailed calculation sheet (full HEP development). The updated internal events HRA analysis (PSA-WF3-01-HR) justified this due to the insignificance of the event. The event did not appear in a single cutset with the screening value. It was judged inconsequential and no calculation sheet was developed. The event remains in the model logic with a screening value (a higher value than would result from more detailed analysis).

In response to this F&O, Waterford 3 added documentation of simulator observations and control room practices during fire scenarios in EC 46718. However, these documentation enhancements did not impact overall HRA results used in the FPRA. For additional details on specific F&O resolutions refer to the updated Attachment V provided with the LAR Supplement.

h. The assessment of internal events PRA HRA actions used in the FPRA is discussed in the Plant Response Model (PRA-W3-05-003 – Section 3.1). The impact of the fire on the action (instruments, increased stress) is also evaluated.

At the start of the evaluation, all actions outside the MCR were assumed failed (unavailable due to fire).

For individual operator actions and for action combinations the increased stress and other performance shaping factors were adjusted from the nominal value presented in the internal events analysis by increasing each failure probability by a factor of 10 (to account for fire related considerations). This methodology is different from the previous revision which more closely resembled NUREG-1921 Appendix C. This updated approach is a more simplified assessment.

Credited HFEs [in-control room actions] were reviewed with respect to the need for instrumentation availability in order to operate equipment or monitor status in relation to procedural decisions. For the WF3 HFE assessment this included the following input signals:

• Steam generator level

• Steam generator pressure

• Pressurizer pressure

• Pressurizer level

This approach is backed up by documented operator interviews that indicate that for all credited actions multiple and diverse indication is available to diagnose and act for modeled HEPs.

<u>RAI PRA 46</u>

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).

<u>Waterford 3 Response</u>
LAR Attachment W has been revised and resubmitted with the Waterford 3 NFPA LAR Supplement. The cited error, RAI request and response are no longer applicable based on the updated model results documented in the Attachment W included in the LAR Supplement.

<u>RAI PRA 47</u>

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 5 and RAB 6). Please explain this discrepancy.

Waterford 3 Response

LAR Attachment W has been revised and resubmitted with the Waterford 3 NFPA LAR Supplement. The cited error, RAI request and response are no longer applicable based on the updated model results documented in the Attachment W included in the LAR Supplement. In the updated results, none of the delta-LERF values exceed the delta-CDF values for the same fire area.

RAI PRA 48

In W3F1-2012-0005, page 5 of 7, SQ 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 (i.e., without the methods Sensitivity Analysis, but with the "new" credit). Please provide updated LAR information if the LAR is no longer accurate.

Waterford 3 Response

LAR Attachment S has been revised and resubmitted with the Waterford 3 NFPA LAR Supplement. In the updated Attachment S, Modifications S1-5 (RAB 6 fire wrap) and S1-14 (removal of secondary combustibles in RAB 27) are credited in the FPRA. The modifications are credited in the analysis as the FPRA assume both modifications are complete. Implementation item S2-21 (Update plant procedures to satisfy the FPRA mission times for Nitrogen accumulators) is also credited in the FPRA analysis.

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

The Waterford 3 FPRA was substantially revised to support the LAR Supplement. The updated FPRA model includes an assessment of both qualitative and quantitative uncertainty. Quantitative uncertainty intervals have been developed for both CDF and LERF. The Summary Report (PRA-W3-05-007 Revision 2) includes a discussion on the sources of uncertainty for each FPRA task (section 4.0 of PRA-W3-05-007). Fire scenario uncertainty is explained for both scoping fire modeling (most fire scenarios) and detailed fire modeling. Quantitative (parametric) uncertainty results are provided for the detailed fire modeling scenarios, but not the scoping fire modeling scenarios. The applicability and qualitative uncertainty of the generic fire model is addressed in the scenario reports.

The updated Transient Summary Report (PRA-W3-05-006T) includes a thorough discussion on the uncertainty associated with the transient fire scenario development.

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

Partial height walls are no longer credited as a complete barrier but are considered with regard to fire radiant effects. RABs 7A, 7B, 7C and 7D have been combined into a single analysis unit – RAB 7. RAB 8 is now also a single analysis unit of what was previously RABs 8A, 8B, and 8C. The updated analysis is now fully compliant with NUREG/CR-6850 guidance. This RAI request and previous response are no longer applicable based on the updated FPRA model, documentation, and analysis methods.

RAI PRA 51

The FPRA peer review noted in LTR-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 QNS-C1.

Waterford 3 Response

The Waterford 3 Fire PRA was substantially revised in development of the LAR Supplement. The updated FPRA analysis includes several MCA scenarios (all scenarios did not screen out). The updated MCA evaluation was also subject to a focused scope peer review (EC 50448). FSS-G6 was evaluated during the focused scope peer reviews and the analysis met Capability Category II. FSS-G5 remains Not Applicable – fire dampers are treated as passive fire barriers with NUREG/CR-6850 failure values applied. This RAI request and previous response are no longer applicable based on the updated FPRA model, documentation, and analysis methods.

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 LAR Supplement Attachment S.

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.*

<u>Waterford 3 Response</u>

a. The FPRA model has been substantially revised. RAB 8C is now merged into RAB 8 (single analysis unit that was RAB 8A, 8B, and 8C). The VFDR list was also reevaluated and a number of items previously included as VFDRs are no longer assessed as such. The following response should satisfy the request even though the request 'as-worded' no longer directly applies to the details of the current/updated analysis. The impact of VFDRs for 125 volt direct current (VDC) control power from DC bus/panel DC-EPDP-B-DC were addressed in the RAB 8 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. Fire impacts to ID-EUPS-MB are addressed in the Fire Risk Evaluation for RAB 8, as ID-EUPS-MB is in the FRANX database and in the FPRA model.

b. In the updated FPRA analysis, there are no Recovery Actions (RAs) associated with EFW valves. Fire damage to EFW valves EFW-229B and EFW-229A is modeled but only the failure of the valve to remain closed (via hot short) is modeled. Most failures would result in the valves failing in the safe open position. This is not considered a PRA-related failure for the FPRA analysis. Document PSA-W3-03-01 contains details on what failures for EFW valves are modeled and which are excluded. There are no Recovery Actions for these valves in Attachment G of the LAR Supplement.

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. There are also a few instances in the FPRA model where AOV related SOV failures are directly modeled. Some AOV related failures due to SOV hot shorts are included in the FPRA model (see PSA-W3-03-01 for additional details).

<u>hjRAI PRA 54</u>

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 original Waterford 3 NFPA 805 LAR submittal contained a typo. The typo ("DCT" should have been "WCT" for the ACCW system) was corrected in the B-3 table that is contained in the LAR Supplement.

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. Additional ACCW inventory is available through the WCT Basin, which may be aligned if the inventory of the Condensate Storage Pool (CSP) is depleted during an event. 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 configuration for the available suction source 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 3 Fire PRA includes scenarios that impact the alignment of WCT to provide EFW suction.

d. Fire impacts on room cooler cables are modeled for some risk important rooms in the Waterford 3 FPRA model. The internal events PRA update (PSA-WF3-01-SC) included an update to the HVAC success criteria, which removed the HVAC requirement to support operation through the mission time for several vital rooms. The MCR HVAC system is one that was added to the model and is now explicitly credited. 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:
   - Diesel Generators
   - Main Control Room

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

[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.

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.

c. <u>ISLOCA Paths and MSOs</u>

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

MCR abandonment was significantly revised in development of the LAR Supplement. The revised MCR abandonment analysis results are now significant contributors to overall fire CDF & LERF. The revised results indicate that MCR abandonment scenarios are now important contributing scenarios and account for nearly 20% of total CDF. The revised analysis included considerations of fire impacts on alternate shutdown capability and all results subject to reasonableness review. This RAI request and previous response are no longer applicable based on the revised FPRA model, methods and results.

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. 480V 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-05 and 7A-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. The LAR Supplement does not contain the statement in question “*variance has no corresponding PRA basic event and by definition has insignificant risk* “. There are several VFDR related components that have no corresponding PRA or FPRA related basic event. The LAR Supplement Attachment C (Table B-3) provides detailed dispositions for non-modeled VFDRs.

b. The revised analysis used to develop the LAR Supplement includes significant revisions to the VFDRs. The listed VFDRs no longer exist in the NFPA 805 supporting analyses.

c. Spurious operation of any of the three charging pumps can challenge the 'RCS Inventory Control' performance goal. The assignment of VFDRs for these components is correct.

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 Fire Risk Evaluations were all revised to support the LAR Supplement. The updated FRE documents do not use the term 'True method' and the process/methodology used to determine the delta risk is explained in detail in each FRE delta risk calculation. The process is also described in detail in Attachment W of the LAR Supplement.

b. The LAR Supplement has no VFDRs with 'open' status.

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.*

<u>Waterford 3 Response</u>

a. The FPRA was substantially revised during the development of the LAR Supplement. The updated FPRA model did not initially credit Recovery Actions. Based on high delta risk results in several areas, a single action to trip RCPs outside the control room was added. The HFE analysis (PRA-W3-05-044) determined that this action has no adverse impact on the FPRA.

b. Fire barrier integrity (which includes derated fire dampers) is explicitly addressed in the Waterford 3 FPRA Multi-Compartment Analysis, PRA-W3-05-005.

   "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 reflected in the multiple compartment assessment. No derated dampers were found to impact the RAs listed in Attachment G.

c. This portion of the RAI request and previous response are no longer applicable. In the revised FPRA to support the LAR Supplement, there are no scenarios with multiple RAs.

d. This portion of the RAI request and previous response are no longer applicable. The two RAs for RAB 6 associated with EFW valves are no longer credited and not included in the LAR Supplement Attachment G.

e. This portion of the RAI request and previous response are no longer applicable. All four RAs listed in LAR Supplement Attachment G are newly developed actions. It is technically the same RA but is listed four times as it is credited for fires in four separate PAUs. The revised Fire Risk Evaluations include credit for the RAs (the addition of the actions was a result of FRE delta risk results).

FIRE MODELING (FM)

RAI FM 01

NFPA 805, Section 2.4.3.3, states: "The PSA [probabilistic safety assessment] approach, methods, and data shall be acceptable to the AHJ [authority having authority] ... " The NRC staff noted that fire modeling comprised the following:

- *The Consolidated Fire Growth and Smoke Transport (CFAST) model was used to calculate control room abandonment times.*
- *The Generic Fire Modeling Treatments approach was used to determine the ZOI in all fire areas throughout plant.*

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). Reference is made to Attachment J, "Fire Modeling V&V," for a discussion of the acceptability of the fire models that were used.

Specifically regarding the acceptability of CFAST for the control room abandonment time study:

a. *Please provide the input files in electronic format for 60 selected CFAST runs that were conducted, i.e., the input files for the cases with the highest HRR in Tables 6-1 through 6-6 in Control Room Abandonment Times Report (Attachment 1 to WSES Fire PRA Fire Scenarios Report, Calculation No. PRA-W3-05-006b).*

b. *Please describe the effect of purge mode ventilation on MCR abandonment.*

c. *During the audit, the NRC staff noted that the ceiling tile thickness in the control room complex is at least 1/2 in. Please explain why a thickness of 1/8-inch was used in the CFAST analysis. Please provide a sensitivity analysis to quantify the effect of using the smaller thickness on control room abandonment times.*

d. *Please provide the basis for the assumption that the fire brigade is expected to arrive within 15 minutes and describe the uncertainty associated with this assumption, discuss possible adverse effects of not meeting this assumption on the results of the FPRA and explain how possible adverse effects will be mitigated.*

e. *Please provide a justification for using average lower bound heat of combustion values and average upper bound yield values (as opposed to the most conservative values) for the cable mix that is present in MCR panels.*

f. *Please provide a gap/sensitivity analysis to demonstrate that the fire growth rates that were used for transient fire scenarios in the control room in lieu of those specified in FAQ-52 lead to more conservative abandonment time estimates or otherwise provide justification for the deviation from the guidelines in FAQ-52.*

g. *During the audit, NRC staff observed numerous combustibles in the equipment area of the control room complex (e.g. a folded table stored between a plastic step stool and moveable stairs, a copier machine and three trash cans close together around a column, etc.). Please provide assurance that the fires involving these combustibles are bounded by the fire scenarios in the equipment area that were considered in the CFAST abandonment time analysis.*

h. *During the audit, NRC staff observed a stack of 18 large plastic containers with personal protective equipment (PPE) (labeled "MSA") in the staff support area of the MCR complex (in the corridor between the control room and HVAC room). Please provide justification for not considering a fire scenario involving these containers in the control* *room abandonment time study or conduct an analysis to assess the effect of this scenario on the FPRA.*

i. Please explain how the results of the sensitivity analysis in Appendix B of the Control Room Abandonment Times Report were used in the FPRA.

Specifically regarding the acceptability of the Generic Fire Modeling Treatments approach:

j. Please explain how the modification to the critical heat flux for a target that is immersed in a thermal plume described in Section 2.4 of the Generic Fire Modeling Treatments document was used in the ZOI determination.

k. Please explain how the Generic Fire Modeling Treatments approach was applied for fires against a wall or in a corner, and describe any additional analysis that may have been performed for wall and corner fires. Please identify the fire areas and scenarios where a location factor of 2 (wall fires) or 4 (corner fires) was used and describe the maximum stand-off distance from a wall or corner within which a fire is considered to be against a wall or in a corner.

l. Provide technical justification to demonstrate that the Generic Fire Modeling Treatments approach as used to determine the ZOI of fires that involve multiple burning items (e.g., an ignition source and an intervening combustible such as a cable tray) is conservative and bounding.

m. Please describe how the flame spread and fire propagation in cable trays and the corresponding HRR of cables was determined. Please explain how the flame spread, fire propagation and HRR estimates affect the ZOI determination and HGL temperature calculations.

n. Please describe the purpose of the Generic Fire Modeling Treatments supplements that were used and explain what affect each of the supplements had in the analyses. Additionally, provide a discussion of the appropriateness and the bases for the acceptability of the methods used in the supplements.

Regarding the acceptability of the PSA approach, methods, and data in general:

o. From the discussion in Section 6.1 of the Fire Scenarios Report (Calculation No. PRA-W3-05-006) it appears that the HRR of transient fires was based on the values in NUREG/CR-6850 for electric motors (98th percentile HRR of 69 kW). It is stated in the Fire Scenarios Report that "the types of transient fires experienced at nuclear power plants were mainly electrical fires." Please provide technical justification for using 69 kW instead of 317 kW for the 98th percentile HRR of transient fires. Please conduct an analysis to quantify the effect of using the higher HRR of 317 kW on the Fire PRA, where the 69 kW cannot be justified.

p. The staff is concerned about the possibility that non-cable intervening combustibles were missed in areas of the plant. Please provide information on how intervening combustibles were identified and accounted for in the fire modeling analyses.

<u>Waterford 3 Response</u>

<u>Part a</u>. The sixty CFAST input files selected during the audit for the Main Control Room Abandonment Calculation were provided via electronic mail (October 11, 2012, from Mr. A. Harris to Mr. M. Janssens). The additional CFAST input files developed to include the smoke purge mode under the most optimistic activation assumptions (documented in PRA-W3-05-026) are available upon request.

Part b. The smoke purge mode for the MCR is designed for smoke removal after the fire has been suppressed and has a low exhaust rate as compared to the normal HVAC flows in the MCR. PRA-W3-05-027 provides an assessment of the effect of the smoke purge mode and it is shown that under the most optimistic assumptions the smoke purge mode could reduce the probability of control room abandonment. However, when considering the time required to manually activate the smoke purge mode, it is concluded in the analysis that the system is not a significant factor in the control room abandonment times.

Nevertheless, for completeness and conformity with the Section 11.5.2.11 of NUREG/CR-6850 guidelines for assessing the control room abandonment times, the MCR abandonment calculation has been updated to include a group of fire scenarios that include the smoke purge mode under the most optimistic activation assumptions as documented in PRA-W3-05-026. Additional discussion is also provided on the flow characteristics and damper alignments when the smoke purge system mode is activated. These new fire scenarios are not used directly in the FPRA but instead provide an indication of the potential benefit in crediting the existing smoke purge system.

Part c. A sensitivity assessment of the ceiling tile thickness is provided in PRA-W3-05-027 for three test cases. The results indicate that increasing the ceiling thickness does not change the results for the two multiple bundle scenarios and only slightly affects the results for the single bundle electrical panel fire scenario considered.

MCR abandonment calculation has been updated in PRA-W3-05-026, to reflect the thicker ceiling tile material using the thermal properties for fiberboard.

Part d. A review of the fire brigade drill times for fire brigade arrival in the control room is summarized in PRA-W3-05-027, which shows that the average arrival time is 9 minutes and the maximum response time is 15 minutes for thirty-five drills conducted between October 14, 2011 and August 14, 2012. Although the drill time data indicate the maximum fire brigade arrival time is fifteen minutes, the MCR abandonment calculation (PRA-W3-05-026) has been updated to include a sensitivity assessment on the assumed fire brigade arrival time for completeness.

Part e. A sensitivity assessment of the calculated abandonment times to the variations in the cable composition (assumed combustion properties) is provided in PRA-W3-05-027. The MCR abandonment calculation PRA-W3-05-026 uses fuel properties for an equal mix of Neoprene, Ethylene Propylene Rubber (EPR), and Hypalon jacketed cables. The fuel properties for individual cables are determined from the most conservative values for each fuel type. Because the average value is used, different mix fractions could produce an effective fuel with more adverse properties relative to visibility. A set of test cases was used in PRA-W3-05-027 to assess this potential and it was determined that a fuel characterized entirely by Neoprene jacketed cables could result in a significant reduction in the predicted abandonment time. However, when viewed in terms of the model bias and model uncertainty, it can be shown that the probability a Neoprene jacketed cable leads to an abandonment time shorter than the baseline case is 7.5×10^{-4} or less. In addition, recent cable fuel data provided in NUREG/CR-7010 indicates that fuel properties obtained from the SFPE Handbook of Fire Protection Engineering, Section 3–4 may be conservative by a factor of four. The use of the newer properties does not appreciably affect the abandonment times or the probability of control room abandonment. It is therefore concluded that the baseline fuel properties as deduced from an equal mix of Hypalon, EPR, and neoprene cable jacket materials are conservative for this application.

Part f. PRA-W3-05-027 provides a sensitivity analysis of the predicted control room abandonment times to the assumed growth rate for transient ignition sources. The analysis shows that the transient ignition source growth rate assumed in the original MCR abandonment calculation may be conservative or non-conservative relative to the guidance specified in NUREG/CR-6850, Supplement 1, depending on the particular NUREG/CR-6850 heat release rate bin considered. It is further concluded that the predicted abandonment time results are somewhat sensitive to the assumed growth rate, but that the conservatism provided by the assumed abandonment temperature bounds the variation in the predicted abandonment times caused by the non-conservative growth rates as found in PRA-W3-05-027. Although the effect of the assumed growth rate is bound by the conservative abandonment criteria assumed, the MCR abandonment calculation (PRA-W3-05-026) has been updated to reflect the recommended transient growth rates in NUREG/CR-6850, Supplement 1.

Part g. The MCR abandonment calculation PRA-W3-05-026 postulates both the fifteen bin transient heat release rate fire scenario and rapidly growing plastic pool fire scenarios in the equipment area. The plastic pool fire scenarios are intended to account for the accumulation of transient material that are not readily characterized by the NUREG/CR-6850 transient heat release rate ignition source. They were included in the MCR abandonment calculation specifically to account for miscellaneous combustible materials identified in this area during the initial survey PRA-W3-05-026. The combustibles observed during the original survey consisted of a small and large plastic step stool (movable stairs). Fire scenarios were developed for each stool as well as for both stools. During the NRC audit, additional combustible materials were observed in this general area, including a folding plastic table between the movable stairs and a proximate photocopier and trash receptacle.

The rapidly growing plastic pool fire scenarios lead to abandonment conditions is less than two minutes for both the large step stool scenario and the scenario with both step stools PRA-W3-05-026. Per Figure 3–11 in PRA-W3-05-026, abandonment is predicted before either fire reaches the calculated peak heat release rate, which means that abandonment is predicted to occur during the growth stage of the fire. Additional mass associated with this fuel package would lead to a larger predicted peak heat release rate, but because the growth rate would remain constant, the predicted abandonment time would not change. Consequently, the rapidly growing plastic pool fire scenarios are broadly applicable to other fuel load configurations and masses due to the short predicted abandonment time.

The fuel package combination that includes the photocopier and the trash container is another fuel package that is not well characterized by the NUREG/CR-6850 transient test data primarily because of the photocopier. The trash container alone is readily characterized by the NUREG/CR-6850 transient ignition source, though the growth rate for this item would be eight minutes rather than two minutes as assumed in the revised MCR analysis PRA-W3-05-026. Based on the NUREG/CR-6850 transient test series (NUREG/CR-6850, Table G-7), the maximum heat release rate expected from a small to medium sized trash container is on the order of 150 kW (142 Btu/s) or less. The heat release rate for a photocopier fire is not easily quantified due to a lack of full scale test data on such commodities; however, the photocopier consists of similar materials and components as computer equipment (computer monitors and desktop towers) for which fire test data is available The peak heat release rate from single computer monitors and desktop towers as reported in the SFPE Handbook of Fire Protection Engineering, Section 3–1 ranges from 250 – 400 kW (237 – 379 Btu/s) with growth times ranging from three to four minutes. The peak heat release rate from a composite trash container-photocopier fire scenario would thus be ~ 550 kW (521 Btu/s), with the peak heat release rate occurring between three and eight minutes after ignition. This fuel package heat release rate is bound by the rapidly growing plastic fuel package fire in terms of the growth rate and peak fire size. The peak heat release rate of the combined fuel package is also lower than the point heat release rate for the Bin 15 transient fuel package, which is 578 kW (548 Btu/s) as presented in PRA-W3-05-027. This means that the transient heat release rate ignition source can be applied to this fuel package combination since the heat release rate range includes the maximum expected heat release rate when full involvement of the fuel package is postulated.

Part h. PRA-W3-05-027 provides a sensitivity assessment of the control room abandonment times on various potential fires involving the Self-Contained Breathing Apparatus (SCBA) equipment in the corridor area for a single ventilation configuration. It is found that although the peak heat release rate for the SCBA fuel package could be significantly higher than the workstation fuel package fire evaluated in the computer room, a similar mechanism exists for smoke and energy transport between the support areas and the MCR proper. The analysis of the SCBA polyethylene fires using the test case configuration indicates that the predicted abandonment in the MCR would be greater than twenty-five minutes, the same result reported for the workstation fires.

The SCBA fire scenarios are included in the MCR abandonment calculation (PRA-W3-05-026) as a baseline cases for each HVAC and natural ventilation configuration evaluated for completeness and consistency with the observed fuel loads in the corridor. The analysis shows that there are some ventilation configurations in which the abandonment time is shorter than twenty five minutes, the baseline value comparable to the computer room fire scenarios; however, the criterion that triggers abandonment is the conservative 50°C (122°F) immersion temperature threshold in all cases. When viewing the results in terms of the NUREG/CR-6850 abandonment criteria, abandonment is not predicted in the MCR for either the computer room fire scenario or the polyethylene SCBA fire scenarios in all cases. In addition, the abandonment time for the polyethylene pool fire in the Equipment Area bounds the SCBA fire scenario in all cases.

Part i. The sensitivity analysis results summarized in Appendix B of Revision 1 of the control room abandonment report (PRA-W3-05-026) are not directly used in the FPRA. They are provided in the abandonment calculation to identify configurations or assumptions that could affect the calculation results with the intent to either demonstrate conservatism in the baseline cases or to provide an application limit. PRA-W3-05-027 provides a detailed assessment of each parameter sensitivity study. The baseline scenarios in the updated MCR abandonment calculation reflect conservative parameter assumptions per Appendix B of PRA-W3-05-026.

Part j. The Generic Fire Modeling Treatments document referenced in this request is no longer used as the source for determining ZOIs. The revised model uses detailed fire modeling methods described in NUREG-1805 utilizing Fire Dynamic Tools (FDTs). The calculations to determine ZOIs for Waterford 3 Fire scenarios are documented in PRA-W3-05-006F and PRA-W3-05-006T. This RAI request and previous response are no longer applicable based on the revised Fire Modeling methods used to support the LAR Supplement.

Part k. The Generic Fire Modeling Treatments document referenced in this request is no longer used as the source for determining ZOIs. The revised model uses detailed fire modeling methods described in NUREG-1805 utilizing Fire Dynamic Tools (FDTs). The increase in transient fire source ZOI near corners and walls is documented in PRA-W3-05-006T. This RAI request and previous response are no longer applicable based on the revised Fire Modeling methods used to support the LAR Supplement.

Part l. The Generic Fire Modeling Treatments document referenced in this request is no longer used for evaluating fires with multiple burning items. This RAI request and previous response are no longer applicable based on the revised Fire Modeling methods used to support the LAR Supplement.

Part m. Flame spread follows the guidance of NUREG/CR-6850 and NUREG/CR-6850, Supplement 1 in terms of characterizing the fire propagation through a vertical cable tray stack and flame spread along cable trays as appropriate for the cable tray configuration of the PAU being analyzed. The impact of flame spread and secondary ignition on HGL temperatures is documented in the revised Multi-Compartment Analysis (PRA-W3-05-005, Revision 1). This analysis was revised to support LAR Supplement development.

Part n. The Generic Fire Modeling Treatments document referenced in this request is no longer used in the Waterford 3 NFPA 805 analysis. The Fire Modeling methods were revised to support development of the LAR Supplement. The revised analysis used more detailed fire modeling methods as opposed to 'generic' ones. This RAI request and previous response are no longer applicable based on the revised Fire Modeling methods used to support the LAR Supplement.

Part o. The Waterford 3 FPRA model was substantially revised to support development of the LAR Supplement. The revised model replaced each 69 kW (65 Btu/s) transient fire (the subject of the request) with a 317 kW (300 Btu/s) transient fire. The revised analysis is now in full compliance with the NUREG/CR-6850 methodology. This request is no longer applicable based on the revised FPRA model, methods, and documentation used to support the LAR Supplement.

Part p. The Waterford 3 FPRA model was substantially revised to support development of the LAR Supplement. The revision included evaluation of non-cable intervening combustibles. Several areas were found to have secondary combustibles and the impacts were evaluated in the appropriate fire scenarios. The revised model and evaluation of such combustibles led to a plant modification to remove a significant amount of combustible materials in RAB 27 (See LAR Supplement Attachment S item S1-14).

Fire Modeling RAI 01.01

By letter dated October 16, 2012, the licensee responded to Fire Modeling (FM) RAI 01.n and described the purpose of the three supplements to the Generic Fire Modeling Treatments that were used in the Fire Probabilistic Risk Assessment (FPRA). The NRC staff has determined it is unclear whether secondary combustibles or panel fire propagation were considered in all of them. For example, the responses discuss whether secondary combustibles were considered in Supplement 2, but not for Supplement 5.

Please state whether detailed fire modeling was required or performed at Waterford 3 to account for secondary combustibles or panel fire propagation and provide the results of such additional analysis. If such detailed fire modeling was not performed, provide a technical justification for why it was not necessary.

Waterford 3 Response

The Waterford 3 Fire Modeling effort was substantially revised to support the development of the LAR Supplement. Detailed fire modeling was conducted in Waterford 3 Physical Analysis Units (PAUs) to address ignition of secondary combustibles and fire propagation between adjacent electrical panels. This detailed fire modeling replaced the previously applied Generic Fire Modeling Treatments.

As part of the transition to a more aligned NUREG/CR-6850 Fire PRA, a re-evaluation of all fire ignition sources in Waterford 3 PAUs was performed. This included additional walkdowns to examine the placement and extent of secondary combustible sources in the PAUs and a re-evaluation of ignition sources to determine if secondary ignition and associated fire expansion could occur (PRA-W3-05-022 – Waterford 3 FPRA Fire Walkdown Report).

The fire development within cable trays is assessed using the methods presented in NUREG/CR-6850. Heat release rates were selected from NUREG/CR-6850, NUREG-1805, or NUREG/CR-7010 as appropriate. The PAUs where secondary ignition was significant were found to involve oil fires or electrical panel ignition sources (PRA-W3-05-006F - Fire Scenarios Report - Fixed Sources).

Walkdowns (PRA-W3-05-022 and PRA-W3-05-028 - MCR Analysis Notebook) were also performed to gather information related to fire propagation between adjacent electrical panels including the panels in the Main Control Room (MCR). The review considered whether panels between cabinets were solid or had cut outs and if air spaces existed between panels. Using this information, panel propagation was reexamined during a comprehensive update to the fixed fire sources using guidance provided in NUREG/CR-6850 Appendix S, Supplement 1 to NUREG/CR-6850 and information from accepted Fire PRA FAQs (Frequently Asked Questions).

Detailed fire modeling was performed as a part of the fire PRA revision. The use of detailed fire modeling by implementing FDT and limited CFAST modeling provided two benefits. First, the use of more detailed fire modeling based on plant specific spatial information allowed for the fire impacts from ignition sources to be reduced. This resulted in lower frequency contributions from individual fire sources which offset the frequency increases caused by the removal of UAMs present in the prior evaluation. Second, it provided more plant-specific insights into the risk significant PAUs.

As a part of the modeling impacts the potential for secondary combustibles ignition and/or panel fire propagation was considered. The inclusion of these aspects identified that specific transient combustible scenarios could result in secondary combustion of fixed combustible loads in RAB 27. Implementation Item S1-14 in Table S-1 of LAR Supplement addresses secondary combustibles in RAB 27. Additional secondary combustible scenarios were also found in other PAUs, including RAB 5, RAB 6, RAB 7, RAB 8, RAB 1 and the TGB (PRA-W3-05-022). Panel fire propagation was limited to those panels without intervening gaps and/or solid panels. Heat and flame effects on adjacent panels were considered along with the potential for hot gas layer generation which was identified for cabinets in RAB 7.

<u>RAI FM 02</u>

NFPA 805, Section 2.5, requires damage thresholds be established to support the performance-based approach. Thermal impact(s) must be considered in determining the potential for thermal damage of structures, systems, or components (SSCs). Appropriate temperature and critical heat flux criteria must be used in the analysis.

It is stated in Section 4.1 on page 4-2 of the Fire Scenarios Report that, "Other than the possible use of some thermo-plastic cables for non-safety network/communications wiring all cables are confirmed as thermoset. Consequently the damage thresholds for thermoset cables as specified in NUREG/6850 are used in this evaluation for determination of ZOIs." Please provide the following information:

a. *Please characterize the installed thermoset and thermoplastic cabling in the power block specifically with regard to the critical damage threshold temperatures and critical heat flux threshold as described in NUREG/CR-6850. Additionally, please provide a statement regarding the extent of installed thermoset cable insulation.*
b. *If thermoplastic cabling is present, discuss the additional targets created/identified using the lower critical temperature damage threshold and/or critical heat flux damage*

threshold criteria of NUREG/CR-6850. In addition, explain how raceways with a mixture of thermoset and thermoplastic cables were treated in terms of damage thresholds.

c. *If thermoplastic cabling is present, please discuss impact on ZOI size due to increased HRR and fire propagation.*

d. *If thermoplastic cabling is present, please discuss self-ignited cables and their impact to additional targets created.*

e. *Please explain if and how the effect of holes in closed raceways on the damage thresholds of cables was accounted for.*

f. *If more targets are identified, please describe the impact to CDF and LERF, as well as the ΔCDF and the ΔLERF for those fire areas affected.*

It is stated in Section 4.2 of the Fire Scenarios Report that "NUREG/CR-6850 recommends failure criteria for solid-state control components of 3 kW/m^2 (versus 11 kW/m^2 for IEEE qualified cable and 6 kW/m^2 for non-IEEE qualified cable) be used for screening purposes. However, given that the enclosure would provide protection to the sensitive internal contents from external fire effects, it is reasonable to apply the same zone of influence established for cable damage. The omission of the credit for the enclosure is judged to offset the non-conservatism of the damage threshold."

g. *Please provide technical justification for using the damage thresholds for cables to establish the ZOI for solid-state control component targets located inside an enclosure*

<u>Waterford 3 Response</u>

<u>Part a</u>.

The design specifications for Waterford 3 cables required IEEE-383 qualification. The materials of construction of the cables are consistent with thermoset performance which was the basis for the determination for the FPRA. Findings from supplemental plant walkdowns following the audit documented in PRA-W3-05-022 support this conclusion based on a sampling of raceway cabling. The walkdown did confirm the existence of a very limited number of cables of a thermoplastic nature installed by a design change. These cables are associated with low voltage signals for multiplexer units and are located in non-safety related raceways in several physical analysis units (PAUs). EC 38344, “Waterford 3 Thermoset/Thermoplastic Cable Report” documents the types of cable insulation used in major applications.

The majority of these cables are located in PAUs where fires are conservatively assumed to result in a complete loss of the PAU and the cable damage threshold is not a critical parameter due to the assumed damage. These PAUs are the turbine building, cable spreading room and RAB 27. Critical damage and heat flux thresholds were calculated for multiple cable types (including thermoplastic) to apply to relevant scenarios. In the remaining PAUs, the cable population in any specific raceway is between 1 and 5 cables which were estimated to have a small impact on any existing analyses. In many cases, the thermoplastic cables are shielded by thermoset cables within the tray which would preclude significant heating to the point of damage.

With respect to vertical burning, the walkdown found that the thermoplastic cables were located primarily in horizontal runs and/or are contained in raceways that were covered. Thus, the potential impact from vertical burning would be limited.

<u>Parts b, c, d, e, and f</u>
The revised evaluation treated the identified thermoplastic cables in a manner consistent with the guidance in NUREG/CR-6850. PRA-W3-05-022 and EC 38344 document the results of the

findings associated with the search for thermoplastic cables that followed the audit (this includes population of cables, arrangement relative to other cables and ignition sources, and qualitative evaluation of potential impacts). The following documents were used to determine factors for fire scenarios as critical heat flux, ZOI, and damage thresholds (and all include considerations for thermoplastic cable type).

PRA-W3-05-013 - Documentation of a Transient Fire Source Zone of Influence Using Fire Dynamic Tools (FDTs)

PRA-W3-05-006F - Fire Scenarios Report (Fixed)

PRA-W3-05-006T - Fire Scenarios Report (Transient)

The results of the model revision are documented in PRA-W3-05-007 Revision 2 and are summarized in the LAR Supplement Attachment W. The results presented include the impacts of all FPRA modeling changes and are not specific to the issues associated with these specific requests.

Part g. The treatment of solid state control components in the FPRA has been revised in development of the LAR Supplement. Expanded ZOIs have been developed to explicitly account for sensitive/solid state components. In the revised analysis no credit is taken in the FPRA for the metal enclosure that may surround the solid state control components. In addition, transient response of the solid state control components is ignored. The treatment in the FPRA is thus conservative and consistent with the NUREG/CR-6850 damage thresholds for screening calculations.

Fire Modeling RAI 02.01

By letter dated October 16, 2012, the licensee responded to FM RAI 02.e and stated that "Holes in closed/sealed raceways were not considered in the FPRA."

Please state whether there are no closed or sealed raceways with holes or the effect of holes in closed/sealed raceways on thermal exposure of cables was ignored and provide justification if such holes were not considered in the FPRA.

Waterford 3 Response

The FPRA supporting the NFPA 805 LAR was substantially revised in development of the LAR Supplement. Detailed walkdowns were performed for all PAUs during this revision process. One aspect of these walkdowns was to confirm the existence or absence of tray covers (above and below) for cable trays and the potential for holes or penetrations in cable tray covers. Particular attention was paid for cable trays potentially exposed to direct flame impingement.

The walkdowns identified areas where top covers were not present primarily due to cables entering the tray from (in many cases) a closed conduit. In some locations non-safety cables (such as the SW corner of RAB 8) were not covered to the ceiling prior to transitioning through the ceiling or floor. Some locations were found with vented covers such as the 4 kV bus duct tray. Although coverage was absent at some locations as noted above, no holes were found in the cable tray covers examined. For safety-related trays the absence of covers was small (judged to be less than 5% of the surface area with the exception of RAB 1E). RAB 1E has many open topped cable trays, but also has a tray-based suppression which would not be possible if covers were present.

In order to determine the status of cable trays covered with HEMYC, cable tray drawings were reviewed to determine if they provided information concerning cable tray covers involving trays wrapped in HEMYC fire wrap. Additionally, plant walkdowns were performed with fire protection staff to visually examine the locations of concern. The location was then discussed with plant fire protection staff to provide their insight into the status of tray covers based on the observed condition. For the cases examined there are two main configurations for the HEMYC wrap.

The first is to provide a fire barrier between cable trays of close proximity. If only partial HEMYC coverage was observed, then it was possible to determine the presence of cable tray covers. If the HEMYC wrap completely covered the cable tray the status was not as clear. However if there were no conduit penetrations associated with the location and the cable tray was covered before and after the HEMYC wrap it was assumed that the tray under the wrap was also covered.

The second usage observed was to provide a barrier at a point on the cable tray where cables either entered or exited the tray. In those cases the HEMYC was centered at the entry/exit point and conduits and/or cable trays would be visible exiting the HEYMC. Based on observation of non-HEMYC covered locations of a similar nature it was assumed that the covers were removed where the cables entered/exited the cable tray.

The walkdown findings were reflected in the revised PAU fire assessment with regard to flame impingement and heat loading when assessing the potential for cable fault performance and when considering cable tray propagation. Cables contained in vented trays or those without bottoms were considered susceptible to direct flame and subsequent secondary ignition. Cables in trays with only bottoms were susceptible to heat loading from below and if ignited were allowed to propagate upward if ignition temperatures were reached. Cables contained in trays with both bottoms and tops (most safety cables) were treated in a manner similar to cables contained in conduits. They were considered failed if their temperature criterion was reached, but would not result in a secondary ignition expanding the original fire ZOI.

As presented, no default assumption as to raceway covers or holes was carried forward in the revised assessment.

RAI FM 03

NFPA 805, Section 2.7.3.2, "Verification and Validation," states: "Each calculational model or numerical method used shall be verified and validated through comparison to test results or comparison to other acceptable models."

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). Reference is made to Attachment J, "Fire Modeling V&V," for a discussion of the verification and validation (V&V) of the fire models that were used.

Furthermore Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805" of the Transition Report states that "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." Regarding the V&V of fire models:

a. *Please describe how the empirical equations/correlations in the Generic Fire Modeling Treatments document and supplements were verified (i.e., how was it ensured that the empirical equations/correlations were coded correctly).*

b. *Please describe the V&V of the empirical equations and correlations identified in the supplements to the Generic Fire Modeling Treatments document and provide assurance that these equations/correlations were applied within their appropriate scopes and limitations.*

c. *Please provide technical details to demonstrate that fire models that are not discussed in Attachment J of the LAR, if any, have been applied within the validated range of input*

> *parameters, or to justify the application of the model outside the validated range reported in NUREG-1824 or other V&V basis documents*

Waterford 3 Response

Part a. The updated Fire Modeling methods used to support the LAR Supplement no longer reference the 'Generic Fire Modeling Treatments' document. The updated analysis has been completed with more detailed fire modeling. This RAI request and previous response are no longer applicable based on the revised Fire Modeling methods used to support the LAR Supplement.

Part b. The updated Fire Modeling methods used to support the LAR Supplement no longer reference the 'Generic Fire Modeling Treatments' document. The updated analysis has been completed with more detailed fire modeling. Attachment J of the LAR Supplement provides the V&V discussion for the fire models applied in the revised assessment.

Part c. Attachment J of the LAR Supplement lists the fire models applied to the Waterford 3 analysis.

RAI FM 04

NFPA 805, Section 2.7.3.3, "Limitations of Use," states: "Acceptable engineering methods and numerical models shall only be used for applications to the extent these methods have been subject to verifications and validation. These engineering methods shall only be applied within the scope, limitations, and assumptions prescribed for that method"

Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," of the Transition Report states that "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."

Regarding the limitations of use, identify uses, if any, of the Generic Fire Modeling Treatments outside the limits of applicability of the method and for those cases, please justify the use of the Generic Fire Modeling Treatments approach or describe the alternate analysis that was conducted.

Waterford 3 Response

The updated Fire Modeling methods used to support the LAR Supplement no longer reference the 'Generic Fire Modeling Treatments' document. The updated analysis has been completed with more detailed fire modeling. Based on the revised methods used to support the Waterford 3 LAR Supplement this RAI request and previous response are no longer applicable.

RAI FM 05

NFPA 805, Section 2.7.3.5, "Uncertainty Analysis," states: "An uncertainty analysis shall be performed to provide reasonable assurance that the performance criteria have been met."

Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," of the Transition Report states 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."*

Regarding the uncertainty analysis for fire modeling:

a. *Please describe how the uncertainty associated with the fire model input parameters (compartment geometry, radiative fraction, etc.) was accounted for.*

b. *Please describe how the "model" and "completeness" uncertainties were accounted for.*

<u>Waterford 3 Response</u>

<u>Parts a & b:</u>

The FPRA analysis was substantially revised in development of the LAR Supplement. The updated analysis has been completed with more detailed fire modeling. The updated fire modeling documentation includes detailed assessments of the uncertainly associated with the analysis. PRA-W3-05-007 Revision 2, the FPRA Summary Report, discusses sources of epistemic and aleatory uncertainty. Uncertainty associated with specific fire modeling parameters is addressed through the use of a conservative and bounding analysis. For more specific and additional details on relevant uncertainties, refer to the 'Assumptions and Uncertainties' and/or the 'Uncertainty' sections of the following supporting calculations:

- PRA-W3-05-013 - Documentation of a Transient Fire Source Zone of Influence Using Fire Dynamic Tools (FDTs)
- PRA-W3-05-006F - Fire Scenarios Report (Fixed)
- PRA-W3-05-006T - Fire Scenarios Report (Transient)
- PRA-W3-05-028 - Waterford 3 Fire PRA Main Control Room Analysis Notebook
- PRA-W3-05-030 - Documentation of Fire Modeling Scenarios for WSES3 PAUs RAB 7A, 7B, 7C, and 7D Hot Gas Layers (using CFAST)
- PRA-W3-05-032 - Uncertainty Assessment for Waterford 3 Fire PRA

<u>RAI FM 06 (Fire Modeling RAI 06)</u>

By letter dated September 27, 2012, the licensee responded to Programmatic RAI 03. In the response, the discussion regarding the qualifications of users of engineering analyses and numerical models was insufficient regarding to fire modeling analyses that were performed during transition.

NFPA 805 Section 2.7.3.4, "Qualification of Users," states: "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."

Please describe what constitutes appropriate qualifications for your staff and consulting engineers that performed the fire modeling analyses during transition and the processes for ensuring their adequate qualification. In addition, please describe how the exchange of information between fire modeling analysts and FPRA personnel was accomplished (e.g., whether the engineers and personnel who performed the fire modeling analyses walked down the fire areas that they analyzed).

Waterford 3 Response

The team performing fire modeling assessments and Fire PRA assessments are qualified through the contractor's qualification card program (RSC 00-01QX, January 2009) which is in compliance with guidance of the National Academy for Nuclear Training. Beyond qualification training in the use of PRA and the performance of fire walkdowns, the individuals have been trained by a recognized industry expert in the use and limitations of the Fire Dynamics Tools (FDT) suite of tools, the use of Fire Dynamic Simulation (FDS), and Consolidated Model of Fire and Smoke Transport (CFAST) results as a part of their overall Fire PRA qualification.

Exchange of information between fire model analysts and Fire PRA personnel was accomplished because the same engineers that performed the fire modeling were involved in the revised Fire PRA plant walkdowns conducted to examine transients, fixed sources, and potential for multi-compartment impacts.

These engineers are also the Fire PRA staff responsible for implementing the fire analysis and PRA impacts involved for the Waterford 3 Fire PRA re-analysis. As PRA analysts, all have been qualified through the contractor qualification card program.

The qualification program is based on an understanding of NUREG/CR 6850 methods and is consistent with the supporting requirements of the ASME PRA standard (ASME/ANS RA-Sa-2009) and Regulatory Guide 1.200.

The process by which Entergy accepts the qualification of the consultants performing the fire modeling analysis is within the contract process and is set forth in EN-DC-156 "Technical and Quality Requirements for Engineering Contracted Services."

RAI FM 07 (Fire Modeling RAI 07)

Several of the first round FM RAI responses make reference to 'forthcoming work.' For this reason, a final review of these RAI responses to determine their adequacy for use by reference in the safety evaluation is not possible. A list of FM RAIs that are affected by this reference to 'forthcoming work' is provided below.

- *Re-analysis of hot gas layer (HGL) assessments for areas where secondary ignition is deemed plausible (RAI 01.m and FM RAI 01.n)*
- *Use of 317 kiloWatts (kW) instead of 69 kW for the heat release rate (HRR) of transient fires (FM RAI 01.o)*
- *Refinement of the probabilistic risk assessment (PRA) to include scenarios with non-cable intervening combustibles that were overlooked (FM RAI 01.p)*
- *Additional analysis to identify thermoplastic cable targets (FM RAI 02.b)*
- *Additional analysis to assess the effect on the ZOI from increased HRR and flame spread of thermoplastic cables (FM RAI 02.c)*
- *Impact of self-ignited cables in the turbine building, cable spreading room, and RAB 27 (FM RAI 02.d)*
- *Risk impact of thermoplastic cables (FM RAI 02.f)*
- *Re-analysis of solid state equipment in vented cabinets (FM RAI 02.g)*

Most of these topics also pertain to FM RAI 04, Limitations of Use, as well as several PRA RAIs.

Please provide the results of this 'forthcoming work' to the NRC staff for final review and describe if any of the final conclusions regarding core damage frequency (CDF), delta (∆) CDF, large early release frequency (LERF), and ∆LERF are changed.

Waterford 3 Response

The Waterford 3 NFPA 805 LAR Supplement provides the summary of the updated results. The results presented in the LAR Supplement are the product of significant changes to the applied Fire PRA and Fire Modeling methodologies (including but not limited to the items listed in this request). Attachment W of the LAR Supplement includes the core damage frequency (CDF), delta (Δ) CDF, large early release frequency (LERF), and ΔLERF that result from the updated model and methodologies.

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) As indicated in the Waterford 3 LAR Supplement, Calculation EC-F13-001 “Post Fire NSCA (NFPA-805)” has been completed and addresses all elements necessary to meet NFPA 805 Section 2.4.2.

b) 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 raceway and by fire area in the Plant.

Safe Shutdown RAI 01.01 (RAI SSA 01.01)

1. By letter dated September 27, 2012, the licensee responded to Safe Shutdown Analysis (SSA) RAI-01(a) and stated, “Updating of the SSA is occurring as each engineering package on each subject is generated.” Analyses for transition to NFPA 805 should be essentially complete at the time the LAR is submitted. It is unclear from the RAI response if all analyses are complete or some are still ongoing. Please provide the following information:

a. A concise description of the state of completion of the analyses that support compliance with the Nuclear Safety Capability Assessment (NSCA) requirements of NFPA 805, Section 2.4.2, with the exception of the Multiple Spurious Operation (MSO) and Non-Power Operation (NPO) analyses, which were stated *as completed in the RAI response. If any analyses necessary to support compliance with NFPA 805 Section 2.4.2 are not complete, please include a description of the remaining work, schedule for completion, and impacts on the information and analyses contained within the LAR and RAI responses. If no impact is expected, please provide a statement and justification for this expectation.*

b. *Please provide the status of the latest post-fire SSA. Revision 3 of the SSA was in draft at the time of the audit. If the latest revision is in draft, please provide the estimated completion date and describe the work that remains to be done.*

2. *By letter dated September 27, 2012, the licensee responded to SSA RAI-01(b), and stated, "Waterford 3 will also be transitioning the Fire Protection Analysis to utilize ARC software." At the time of the audit, licensee staff associated with the NSCA development indicated that the SSA would remain the governing analysis documentation for demonstrating compliance and there was no specific commitment to use the ARC software. Please provide the following information:*

a. *A discussion of the use of the ARC software in demonstrating compliance with NFPA 805, the continuing role of the SSA and the integration of the two.*

b. *A discussion of the databases and software that integrate fire protection program structure, system, and component data; fire modeling results, and PRA analyses (e.g., EPM-SAFE-PB and ARC) having a range of uses applicable to NFPA 805 implementation. These uses are subject to several NFPA 805 requirements including those that address determination of success paths; completion of the NSCA; the quality, configuration control, documentation, and verification and validation of analyses; and limitations of use. In addition, these databases and software can be used to facilitate integration of several aspects of NFPA 805 compliance. Specific applicable NFPA 805 requirements include:*

NFPA 805 Section 2.2.9, "Plant Change Evaluation," states that: "In the event of a change to a previously approved fire protection program element, a risk informed plant change evaluation shall be performed and the results used as described in 2.4.4 to ensure that the public risk associated with fire-induced nuclear fuel damage accidents is low and that adequate defense-in-depth and safety margins are maintained."

NFPA 805, Section 2.2.11, "Documentation and Design Configuration Control," requires that: "The fire protection program documentation shall be developed and maintained in such a manner that facility design and procedural changes that could affect the fire protection engineering analysis assumptions can be identified and analyzed."

NFPA 805 Section 2.4.1, "Fire Modeling Calculations," requires: (2.4.1.1) "The fire modeling process shall be permitted to be used to examine the impact of the different fire scenarios against the performance criteria under consideration." (2.4.1.2.1) "Only fire models that are acceptable to the authority having jurisdiction shall be used in fire modeling calculations." (2.4.1.2.2) "Fire models shall only be applied within the limitations of that fire model." (2.4.1.2.3) "The fire models shall be verified and validated."

NFPA 805 Section 2.4.3.3 regarding fire risk evaluations states: "The PSA [probabilistic safety assessment] approach, methods, and data shall be acceptable to the AHJ [authority having jurisdiction]. 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 2.4.4, "Plant Change Evaluation," states: "A plant change evaluation shall be performed to ensure that a change to a previously approved fire protection program element is acceptable. The evaluation process shall consist of an integrated assessment of the acceptability of risk, defense-in-depth, and safety margins. The impact of the proposed change shall be monitored."

NFPA 805 content requirements include:

(2.7.1.1) "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."

(2.7.1.2) "A fire protection program design basis document shall be established based on those documents, analyses, engineering evaluations, calculations, and so forth that define the fire protection design basis for the plant. As a minimum, this document shall include fire hazards identification and nuclear safety capability assessment, on a fire area basis, for all fire areas that could affect the nuclear safety or radioactive release performance criteria defined in Chapter 1."

(2.7.1.3) "Detailed information used to develop and support the principal document shall be referenced as separate documents if not included in the principal document."

NFPA 805 configuration control requirements include:

(2.7.2.1) "The design basis document shall be maintained up-to-date as a controlled document. Changes affecting the design, operation, or maintenance of the plant shall be reviewed to determine if these changes impact the fire protection program documentation."

(2.7.2.2) "Detailed supporting information shall be retrievable records. Records shall be revised as needed to maintain the principal documentation up-to-date."

Finally, NFPA 805 quality requirements apply to use of integration databases and software:

> *(2.7.3.1) "Each analysis, calculation, or evaluation performed shall be independently reviewed."*
>
> *(2.7.3.2) "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) "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) "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." "An uncertainty analysis shall be performed to provide reasonable assurance that the performance criteria have been met."*

The NRC staff notes that, given the broad range of requirements applicable to use of integration databases and software, the Transition Report provided insufficient details for the staff to complete its review of the various areas affected by this software and is requesting that the following additional information be provided.

1) *A description of how the post transition change evaluation process will ensure that the potential interfaces between integration databases and software and other databases and analyses (e.g., the cable and raceway database, the NSCA, the FPRA, and fire modeling) are evaluated and updated, as appropriate.*

2) *A description of the process that will be employed to ensure that integration databases and software are maintained in accordance with documentation and design configuration control processes and procedures.*

3) *A description of the process and procedures that will be used to ensure that integration databases and software analyses are conducted and/or updated by persons properly trained and experienced in its use.*

4) *A description of the processes and procedures that will be used to ensure that integration databases and software analyses comply with NFPA 805 fire modeling, content, and quality control requirements.*

c. *The SSA RAI-01(b) response further states, "Attachment S, item S2-13, implements the actions above." This item describes development of new procedures and processes based on the NSCA analyses, but does not appear to* *address either the specific update of the SSA, or the transition to the use of the ARC software as described in the response.*

Please provide a new implementation item in Attachment S or revise the existing implementation items in Attachment S that specifically address the work to be completed as part of transition, as your response to this RAI or the previous SSA RAI-01.

Waterford 3 Response

1. *Please provide the following information:*

a. ***A concise description of the state of completion of the analyses that support compliance with the Nuclear Safety Capability Assessment (NSCA) requirements of NFPA 805, Section 2.4.2, with the exception of the Multiple Spurious Operation (MSO) and Non-Power Operation (NPO) analyses, which were stated as completed in the RAI response. If any analyses necessary to support compliance with NFPA 805 Section 2.4.2 are not complete, please include a description of the remaining work, schedule for completion, and impacts on the information and analyses contained within the LAR and RAI responses. If no impact is expected, please provide a statement and justification for this expectation.***

NFPA 805 Section 2.4.2 states the following steps shall be performed:
(1) Selection of systems and equipment and their inter relationships necessary to achieve the nuclear safety performance criteria in Chapter 1.
(2) Selection of cables necessary to achieve the nuclear safety performance criteria in Chapter 1.
(3) Identification of the location of nuclear safety equipment and cables.
(4) Assessment of the ability to achieve the nuclear safety performance criteria given a fire in each fire area.

Waterford 3 analyses comply with all of the steps given in NFPA 805 Step 2.4.2. Calculation EC-F13-001, "Post Fire NSCA (NFPA-805)" provides the systems and equipment and their inter-relationships necessary to achieve the nuclear safety performance criteria. EC-F13-001 provides the assessment of the ability to achieve the nuclear safety performance criteria given a fire in each fire area. The cables and their location that support the systems and equipment to achieve the nuclear safety performance criteria are also documented in Calculation EC-F13-001.

b*. Please provide the status of the latest post-fire SSA. Revision 3 of the SSA was in draft at the time of the audit. If the latest revision is in draft, please provide the estimated completion date and describe the work that remains to be done.*

As indicated in the Waterford 3 LAR Supplement, Calculation EC-F13-001 "Post Fire NSCA (NFPA-805)" has been completed and addresses all elements necessary to meet NFPA 805 Section 2.4.2.

2. … *Please provide the following information:*

a. ***A discussion of the use of the ARC software in demonstrating compliance with NFPA 805, the continuing role of the SSA and the integration of the two****.*

As indicated in the Waterford 3 LAR Supplement, Calculation EC-F13-001 “Post Fire NSCA (NFPA-805)” has been completed. Any changes to the plant-approved NSCA will be governed by the site’s configuration control procedures, such as EN-DC-115, “Engineering Change Process” and its reference procedures. These governing processes are planned to be revised for compliance to NFPA 805 and are being tracked by implementation item S2-12.

The ARC software being proposed for use to assist in administering the NFPA 805 Fire Protection Program will be governed by procedure EN-IT-104, “Software Quality Assurance Program”. The ARC software will contain NSCA information and will be used as a tool following NFPA 805 transition to initially evaluate plant changes (i.e., “What-If”) that may potentially impact Waterford 3’s approved NSCA.

b. ***The Transition Report provided insufficient details for the staff to complete its review of the various areas affected by this software and is requesting that the following additional information be provided.***

   1. ***A description of how the post transition change evaluation process will ensure that the potential interfaces between integration databases and software and other databases and analyses (e.g., the cable and raceway database, the NSCA, the FPRA, and fire modeling) are evaluated and updated, as appropriate.***

   2. ***A description of the process that will be employed to ensure that integration databases and software are maintained in accordance with documentation and design configuration control processes and procedures.***

   3. ***A description of the process and procedures that will be used to ensure that integration databases and software analyses are conducted and/or updated by persons properly trained and experienced in its use.***

   4. ***A description of the processes and procedures that will be used to ensure that integration databases and software analyses comply with NFPA 805 fire modeling, content, and quality control requirements.***

PRA Calculations and databases are administered through the PRA governing procedure EN-DC-151 “PSA Maintenance and Update” which specifies how the PRA model, including Fire PRA model, are maintained and updated. The software being used is being maintained using the EN-IT-104 “Software Quality Assurance Program” procedure. The implementation of new software as well as updates/revisions of software must be administered via this procedure. Quality assurance of Fire PRA software (FRANX, CAFTA) is documented in code document packages (CDPs). The level of quality of these programs is equal to that of other Regulatory Commitment software.

Any changes to plant documents are governed by the site’s configuration process through a series of fleet procedures as listed below. These procedures detail how the quality of both the inputs to the software and integration databases (e.g., ARC, cable and raceway database (PDMS), the NSCA, Fire PRA, Fire Modeling) will be maintained.

a. EN-IT-104 Software Quality Assurance Program
b. EN-DC-105 Configuration Management
c. EN-DC-115 Engineering Change Process
d. EN-DC-126 Engineering Calculation Process

e. EN-DC-128 Fire Protection Impact Reviews
f. EN-DC-132 Control of Engineering Documents
g. EN-DC-134 Design Verification
h. EN-DC-141 Design Inputs
i. EN-DC-151 PSA Maintenance and Update
j. EN-DC-179 Preparation of Fire Protection Engineering Evaluations

These procedures have and/or will be updated as necessary to comply with any NFPA-805 requirements (e.g., fire modeling, content, and quality control requirements) and are being tracked by implementation item S2-12. Procedure EN-TQ-212, "Conduct of Training and Qualification" ensures that personnel are properly qualified in the design configuration control processes above.

c. ***The SSA RAI-01(b) response further states, "Attachment S, item S2-13, implements the actions above." This item describes development of new procedures and processes based on the NSCA analyses, but does not appear to address either the specific update of the SSA, or the transition to the use of the ARC software as described in the response. Please provide a new implementation item in Attachment S or revise the existing implementation items in Attachment S that specifically address the work to be completed as part of transition, as your response to this RAI or the previous SSA RAI-01.***

Attachment S, item S2-13 stated, "Several NFPA 805 document types such as: NSCA Supporting Information, Non-Power Mode NSCA Treatment, etc., generally require new control procedures and processes to be developed since they are new documents and databases created as a result of the transition to NFPA 805. The new procedures will be modeled after the existing processes for similar types of documents and databases. System level design basis documents will be revised to reflect the NFPA 805 role that the system components now play."

Attachment S, item S2-13 has been clarified based on the 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 2005 (ADAMS Accession No. MLO50310295). 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. ML091770265), 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 NEI 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 required the Safe Shutdown Analysis (NSCA) to be updated:

- Waterford 3 evaluated rising stem valves included in the NSCA Safe Shutdown Equipment List (SSEL) and concluded that Waterford 3 does not have any recovery actions that would require manual opening or closing of a rising stem valve in order to place the plant in a safe and stable configuration.
- The required for hot shutdown / important for safe shutdown guidance has been incorporated into the NSCA. Waterford 3 treats everything as “required”.
- IN92-18 valve assessments are complete with final resolution listed in Attachment S, items S1-1 and S1-15
- The new criteria introduced in section 3.5.1.1 for evaluating inter-cable hot shorts for proper polarity DC circuits was included during the development of the NSCA. Circuit breaker coordination calculations were 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 was confirmed in addition to the proper alignment of the time-current characteristic curves.
- PRA utilized revision 1 of NEI 00-01 for their analysis. Review indicates no impact on the PRA from revision 2 criteria.

In summary, the Nuclear Safety Capability Assessment (NSCA) and the PRA have been updated to include the additional requirements introduced by revision 2 of NEI 00-01.

Safe Shutdown RAI 02.01 (RAI SSA 02.01)

By letter dated September 27, 2012, the licensee responded to SSA RAI-02, and identified specific gaps between Revision 1 and Revision 2 of Nuclear Energy Institute (NEI) 00-01 as applicable to Waterford 3. NRC staff review of the response has identified the following concerns requiring additional information:

a. *The bulleted responses identify a number of commitments to perform additional work but there is no discussion on how this work may impact the existing analyses, including those analyses that have been reviewed by the staff and may be relied on for approval in the safety evaluation. Please discuss the expected impact of the additional work on the conclusions of the existing analyses (i.e., the nuclear safety capability assessment (NSCA) and Fire PRA, as reviewed by the staff during the audit).*

b. *The second bullet addresses the categorization of safe shutdown components as required for safe shutdown and important to safe shutdown. This equipment categorization aspect of NEI 00-01, “Guidance for Post Fire Safe Shutdown Circuit Analysis, Rev. 2, is not applicable under NFPA 805. As described in NEI 00-01, Rev. 2, the categorization or segregation of components as “required for” or “important to” safe shutdown is associated with the multiple spurious* *operation (MSO) methods there-in. The MSO methods applicable to NFPA 805, as implemented in the LAR, are those associated with the expert panel process as described in NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)", Rev. 2, (ADAMS Accession No. ML081130188), as endorsed by Section 3.3 of Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Rev. 1, 2009 (ADAMS Accession No. ML092730314), and supplemented by NRC FAQ 07-0038, Lessons Learned on Multiple Spurious Operations, (ADAMS Accession No. ML110140242). Please clarify how Waterford 3 intends to use the equipment categorization of NEI 00-01, Rev. 2, and identify the impact on the submitted safe shutdown analyses.*

c. Related to Item a. above, the last paragraph of the response states, "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." Provide the new Attachment S showing the implementation item.

Waterford 3 Response

a) *Please discuss the expected impact of the additional work on the conclusions of the existing analyses (i.e., the nuclear safety capability assessment (NSCA) and FPRA, as reviewed by the staff during the audit).*

There is no expected impact to the documents reviewed by the NRC staff during their audit as a result of Waterford 3 alignment with NEI 00-01, Section 3.0 Rev 2. The following items i) through vi) address the bulleted responses provided to RAI SSA 02.

i) Fire exposure impact on manually operated rising stem valves.

Waterford 3 evaluated rising stem valves included in the NSCA Safe Shutdown Equipment List (SSEL) and concluded that Waterford 3 does not have any recovery actions that would require manual opening or closing of a rising stem valve in order to place the plant in a safe and stable configuration.

ii) Required for Hot Shutdown versus Important to Hot Shutdown guidance incorporated into NSCA (EC 40610)

During development of the Waterford 3 Safe Shutdown Analysis (ECF00-026, Rev 3) there was no distinction made between the components that were determined to be "Required for Hot Shutdown Components" and components which could be classified as "Important to Safe Shutdown Components" as detailed in NEI 00-01, Rev 2, Section 3.4. The same methodology of analyzing each safe shutdown component as being a "Required for Hot Shutdown Component" was implemented during development of the Waterford 3 Nuclear Safety Capability Assessment (NSCA). Therefore, no impact is expected to the NSCA.

iii) Information Notice (IN) 92-18 valve assessment (EC 40636 & WF3-FP-13-00001)

As stated in the previous response to RAI SSA 02, the IN 92-18 valve assessment was completed and required modifications are provided in Attachment S, items S1-1 and S1-15.

iv) Inter-cable hot short for proper polarity DC circuits (EC 41765)

EC 41765 evaluates whether two concurrent hot shorts of the proper polarity, e.g. plus-to-plus and minus-to-minus, within the same multi-conductor cable for DC circuits for credited post-fire safe shutdown components could have negative consequences in the ability to achieve safe shutdown. This evaluation is performed in every fire area through which the cable is routed and concluded that there are no new adverse situations.

v) Circuit breaker coordination for internal breaker tripping devices

Circuit breaker coordination calculations ECE91-055, ECE91-056 and ECE91-253 document Waterford 3's ability to ensure that the protective device nearest the fault will operate prior to the operation of any "upstream" protective devices, thereby limiting the interruption of electrical supply. The above referenced calculations indicate that adequate coordination exists for the credited AC buses at all voltage levels.

For non-safe shutdown loads that require 125 VDC power to ensure the breaker trip capabilities, control power is available to isolate faults on power cables to non-safe shutdown loads.

Based on these determinations, the existing analyses (i.e. NSCA and Fire PRA as reviewed by the staff during the audit) remain valid.

vi) Impact on PRA due to Revision 2 of NEI 00-01

There is no expected impact on the Waterford 3 PRA due to Rev 2 of NEI 00-01.

NEI 00-01 addresses the assessment of circuit analysis post fire. There are two areas where the PRA is involved. The first involves the development of the assessment list as defined in Appendix F utilizing PRA risk ranking measures. The process of identification utilizing the internal events model would not be impacted by the revisions to NEI 00-01. The second area involves the implementation of the findings from the circuit assessment into the fire PRA model to reflect the potential for circuit failure due to fire effects such as multiple spurious operations. The current fire PRA model addresses those failures identified by the existing circuit failure analysis.

Documentation of the reviews and evaluations for the bulleted responses are being prepared in accordance with the Entergy Engineering Change (EC) Evaluation process (procedure EN-DC-115). In the event that a change does occur to a document that was critical in establishing the Waterford 3 NFPA 805 fire protection program and was previously reviewed by the NRC staff, the revised document will be submitted to NRC by means of the Licensing Basis Document Change Process (EN-LI-113). Attachment S has been revised to update engineering and PRA documentation to specifically address items i) through vi) above to indicate that the plant fire protection analysis meets the criteria in NEI 00-01, Rev 2.

Based on the extent of the evaluation to date, Waterford 3 does not expect or anticipate that the results of the evaluations identified above will alter the conclusions of the existing analysis as reviewed by the staff during the audit.

b) ***Please clarify how Waterford 3 intends to use the equipment categorization of NEI 00-01, Rev. 2, and identify the impact on the submitted safe shutdown analyses.***

See response to RAI SSA 02.01 a(ii) above.

c) Provide the new Attachment S showing the implementation item.

Implementation item S2-19 has been added to the S-2 Implementation table to cover the updating of the NSCA to address the transition to NEI 00-01, Revision 2 Section 3.

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

The fragmented sentence no longer applies and was removed from the B-2 Table in the LAR Supplement (See W3F1-2013-0048).

Safe Shutdown RAI 03.01 (RAI SSA 03.01)

By letter dated September 27, 2012, the licensee responded to SSA RAI-03 and stated, "Since the submittal of the LAR, the instrument air compressors have been added to the SSEL [Safe Shutdown Equipment List] and the circuits routed." Please provide an explanation of the basis for this change, including a detailed discussion of the impact of this change on the NFPA 805-related analyses (e.g., NSCA, FPRA, and NPO) and the information previously submitted in the LAR. In addition, should Waterford 3 intend to credit the availability of instrument air, please provide the analysis or justification that the associated instrument air piping and tubing, which may have brazed or soldered joints, will remain free of fire damage.

Waterford 3 Response

The Instrument Air (IA) Compressors were added to the Appendix R CSD SSEL for valves SI-129A(B). The NSCA analysis credits the Nitrogen Gas Backup Air System for air operated valves. No credit is taken in the NSCA analysis for Instrument Air or the Instrument Air Compressors.

Since Waterford 3 is not taking credit for Instrument Air supply to NSCA valves, no analysis is required for the instrument air piping and tubing.

RAI SSA 04

For several of the entries in Table B-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

The reference to the RAIs from the HNP and ONS nuclear plants was removed from the B-2 Table in the LAR Supplement (See W3F1-2013-0048).

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 B-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

The RAI request and previous response are no longer applicable as the LAR Supplement Attachment B (NEI 04-02 Table B-2) element 3.5.1.1 Alignment Basis addresses the DC Circuit Proper Polarity review performed in EC 41765 (DC Circuit Proper Polarity).

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 RAI request and previous response are no longer applicable as the revised analyses that support the LAR Supplement resulted in revisions to Attachment G that removed the Recovery Actions (RAs) cited in the RAI and deletion of Item S1-6 in Attachment S, Plant Modifications. (See W3F1-2013-0048).

RAI SSA 07

Attachment G of the LAR defines the PCS to include the Remote Shutdown Panel Room (LCP-43) and: 1) operation of transfer switches, 2) operation of isolation switches, 3) operation of local control panel switches, 4) operation of power distribution panel switches in the relay or switchgear rooms, 5) manual operation of breakers in the switchgear rooms, and 6) opening of *the battery room doors. Attachment G states that Waterford followed the criteria of FAQ 07-0030, Revision 5.*

The FAQ provides the following criteria for determining whether actions are considered part of the PCS:

> *...actions that are necessary to activate or switch over to a primary control station(s) may be considered as taking place at primary control station(s) under the following conditions:*
>
> *The actions are limited to those necessary to activate, turn on, power up, transfer control or indication, or otherwise enable the primary control station(s) and make it capable of fulfilling its intended function following a fire. These actions must be related to the alternative/dedicated shutdown function and should take place in locations common to panels that perform the transfer of control. For example, switches that disable equipment in order to allow the alternative/dedicated shutdown location to function would be included as part of the primary control station. However, these actions must be in the same location(s) (panel or the local vicinity surrounding the panel) as the normal/isolation switches and may include de-energization of selected equipment and/or circuits (if such actions are similar to the use of isolation switches). This does not include additional actions in the plant that, while necessary to achieve the NSPC, are not part of enabling the primary control station(s) (e.g., controlling inventory by locally controlling valve(s)).*

Not all of the PCS actions described in Attachment G appear to be consistent with the guidance of FAQ 07-0030 regarding those actions necessary to enable the primary control station(s). Therefore, the LAR may not identify and quantify the risk of all RAs. For example, opening the battery room doors does not appear to be a required action for enabling the PCS.

a. For control room evacuation scenarios; please identify which PCS actions in Table G-1, are associated with enabling the PCS, Panel LCP-43.

b. Please provide additional detailed discussion and justification that these actions are necessary to enable the PCS, per the guidance in FAQ 07-0030.

c. If any of these PCS actions are determined to be RAs, then Please provide a positive conforming statement that these RAs are feasible and have been evaluated for risk.

<u>Waterford 3 Response</u>

The RAI request and previous response are no longer applicable. The list of PCS actions in Attachment G has been updated in the LAR Supplement to only include the operation of Fire Isolation and Transfer switches located in the Auxiliary Panel Room. The actions are necessary to activate and transfer control to Waterford's Alternate Shutdown Panel, LCP-43, during a control room fire requiring evacuation and therefore meet the definition of PCS actions per FAQ 07-0030. (See W3F1-2013-0048).

<u>RAI SSA 08</u>

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.*

<u>Waterford 3 Response</u>

a) The analyses that support the LAR Supplement resulted in revisions to Attachment G which explicitly states alignment with the feasibility criteria in FAQ-07-0030. Attachment S, Item S2-17 ensures the training process includes drills associated with recovery actions.

b) Engineering Report WF3-FP-13-00003 includes the feasibility review for the recovery actions contained in Attachment G of the LAR Supplement (see W3F1-2013-0048).

<u>Safe Shutdown RAI 08.01 (RAI SSA 08.01)</u>

1. *By letter dated September 27, 2012, the licensee responded to SSA RAI-08(a) and described how each of the 11 feasibility criteria in FAQ 07-0030, "Establishing Recovery Actions" (ADAMS Accession No. ML110070485), is addressed. It appears that the basis is for compliance with Appendix R and does not yet incorporate the FAQ criteria for demonstrating feasibility to meet NFPA 805. Please provide the following additional information:*

    a. *FAQ 07-0030, Criterion 2, addresses consideration of the availability of systems and indications essential to performing recovery actions (RAs). The letter dated September 27, 2012, references the conclusions of the SSA for plant monitoring instrumentation. The response does not address the availability of systems and indications with respect to the feasibility of performing the RAs identified in the LAR. Please provide a confirmation that the selected plant monitoring instrumentation bounded those systems and indications essential to performing RAs, or provide additional discussion of how Criteria 2 of FAQ 07-0030 is met with regard to determining the availability of those systems and indications necessary to perform the RAs in LAR Attachment G.*

    b. *FAQ 07-0030, Criterion 4, addresses emergency lighting. The letter dated September 27, 2012, states that sufficient emergency lighting is installed to support access/egress to local equipment for required hot standby manual actions. LAR Attachment G and Attachment S, however, state a modification is necessary to install emergency lights. Please clarify the apparent discrepancy between the feasibility analysis as described in the RAI response and the LAR statements that additional lighting is needed. Also, please clarify if emergency lighting is provided at the local equipment to support the performance of the RAs.*

    c. *FAQ 07-0030, Criterion 5, addresses the availability of tools, equipment, and keys required for the RA. The letter dated September 27, 2012, only addresses cold shutdown repairs. Please provide additional discussion of tools, equipment, keys, or any other similar operator aids necessary to achieve the RAs in LAR Attachment G.*

    d. *Please discuss any ongoing or completed actions to incorporate the FAQ 07-0030 criteria in the licensee's documentation for compliance with NFPA 805.*

2. *By letter dated September 27, 2012, the licensee responded to SSA RAI-08(b) and stated that the actions covered in the its feasibility analysis include RAs to meet NFPA 805 Safe and Stable Hot Shutdown. However, based on the NRC staff's review of RAs described in LAR Attachment G for Fire Area RAB-1, there are RAs that do not appear to have been addressed in the feasibility analysis as stated.*

    a. *Please justify the differences between the list of RAs in LAR Attachment G and those listed in the feasibility analysis.*

    b. *Please justify not performing the feasibility analysis on any NFPA 805 RA identified in LAR Attachment G, utilizing the 11 criteria of FAQ 07-0030, if applicable.*

    c. *Please update the SSA if new RAs have been added to meet NFPA 805 Safe and Stable Hot Shutdown.*

<u>Waterford 3 Response</u>

The RAI request and previous response are no longer applicable. The list of recovery actions in Attachment G has been updated in the LAR Supplement (See W3F1-2013-0048) to only require securing the Reactor Coolant Pumps (RCPs) at the local breakers in the event of a fire in Fire Areas RAB1, RAB 7, RAB 8 or the TGB. The securing of the RCPs at their local breakers was confirmed to be feasible based on the satisfying the 11 criteria provided in FAQ 07-0030. The results are documented in Engineering Report WF3-FP-13-0003.

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.

<u>Waterford 3 Response</u>

The RAI request and previous response are no longer applicable. The reference to performance based areas for Fire Areas RCB, Roof E and Roof W was removed from the B-3 Table in the LAR Supplement (See W3F1-2013-0048).

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.

<u>Waterford 3 Response</u>

Waterford 3 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 fire watches 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 Supplement Attachment S, item S2-17, implements the above actions.

Safe Shutdown RAI 10.01 (RAI SSA 10.01)

By letter dated September 27, 2012, the licensee responded to SSA RAI-10 and stated, "Should a recovery action be utilized as a means of reducing fire risk during a NPO High Risk Evolution (HRE), 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 with the time frame required."

Since the licensee is not performing the feasibility criteria per FAQ 07-0030, please describe how the equipment is ensured to be functional and how the operators will be able to feasibly perform the actions within the required timeframe using the criteria of NEI 04-02. Please describe and provide a justification for each of the feasibility criteria in NEI 04-02 that are not applied to NPO RAs, and describe any additional assumptions and criteria that are not prescribed in NEI 04-02 (if there are any).

Waterford 3 Response

Waterford 3 does not expect to have any recovery actions used as a means of reducing fire risk during a NPO higher risk evolution. However, in the unlikely event that NPO recovery actions become desirable, then the endorsed feasibility criteria at that time (FAQ 07-030, NEI 04-02, etc) will be used to ensure equipment is functional and the operator actions can be feasibly performed within the necessary timeframe.

In conjunction with response to RAI SSA 12, Implementation Item S2-17 is revised to ensure that all feasibility criteria are addressed for all Recovery 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.

Safe Shutdown RAI 12 (RAI SSA 12)

By letter dated September 27, 2012, the licensee responded to SSA RAI-06 and stated that when new RAs are implemented (Attachment S, Line Item S1-6), FAQ-07-0030 Revision 5 (11 feasibility criteria) will again be reviewed to verify that the RAs are feasible. The RAI response to SSA RAI 07(c) also states that additional actions determined to be RAs as a result of the updated fire risk evaluation will also be evaluated for risk and feasibility.

Based on the responses provided to SSA RAI-06 and SSA RAI-07(c), please provide a new LAR Attachment S reflecting the commitments to perform the cited feasibility analyses, reviews, and include these actions within the scope of Implementation Item S2-17.

Waterford 3 Response

Item S2-17 in Attachment S has been revised to ensure that all feasibility criteria in FAQ 07-0030 are addressed for NSCA recovery actions. The feasibility and risk of recovery actions as provided in the LAR Supplement are documented in Engineering Report WF3-FP-13-0003.

Safe Shutdown RAI 13 (RAI SSA 13)

By letter dated September 27, 2012, the licensee responded to PRA RAI 57. The NRC staff's review of that response and efforts associated with development of the draft safety evaluation has identified a need for additional information regarding the treatment of RAs:

a. The response to PRA RAI 57, Item c.i, states, "There is a separation issue in the area and fires impacting both A and B charging pumps are possible. The train B pump variance from deterministic requirement (VFDR) is based solely on a credited SSA RA (one that is not included in LAR Attachment G)." If an RA is "credited" to address a separation issue and resolve a VFDR, it should be included in LAR Attachment C, Attachment G, and the additional risk of the action provided in Attachment W. Please provide additional explanation of the credited SSA RA and why this action is not included in the LAR.

b. *Attachment G, Step 2, "Results," states, "The results in Table G-1 identify four RAs (Fire Areas RAB 5 and RAB 6) necessary to meet the risk acceptance criteria. The remaining RAs (Fire Area RAB 1) are required to maintain a sufficient level of defense-in-depth [(DID)]." Contrary to the statements in Attachment G, the Table B-3 summary for Fire Areas RAB 5 and RAB 6 state that, "The fire risk evaluation determined that the applicable risk, defense-in-depth, and safety margin criteria were satisfied without further action." The summary further states under "DID Maintained" that "recovery actions are required for this area to meet defense-in-depth criteria." Please provide the following information:*

   1) *Clarify if the RAs in RAB 5 and RAB 6 are required to meet risk acceptance criteria as stated in Attachment G or defense-in-depth (DID) as stated in Attachment C of the LAR.*

   2) *VFDR 5-13 is listed in Attachment G with the action in RAB 5, but the VFDR disposition (in Attachment C) does not identify that an RA is necessary.*

   3) *VFDRs 6-09 and 6-10 are listed in Attachment G with an action in RAB 6, but the VFDR dispositions (in Attachment C) do not identify that an RA is necessary.*

   4) *Attachment G states that RAB 1 RAs are DID. VFDR 1-054 states, "This variance is identified as an RA action in Attachment G." Is this action required or DID?*

   5) *Attachment W reports the additional risk of RAs for RAB 1, 5, and 6. If all RAs are DID, how was the risk of the actions calculated?*

c. *Attachment C contains a "Fire Risk Summary" for each performance-based area, and each summary includes the following statement: "The fire risk evaluation determined that the applicable risk, defense-in-depth, and safety margin criteria were satisfied without further action" [emphasis added]. However, under the "DID Maintained" heading for several fire areas (e.g., RAB 1, RAB 2, RAB 5, RAB 6, RAB 7A, RAB 7B, RAB 8A-C, RAB 23, and RAB 25) modifications or DID RAs are identified. These modifications and actions are tied to VFDR resolutions associated with these fire areas. Please provide additional explanation to rectify the apparent contradiction of the summary statement that criteria were satisfied without further actions and the identification of RAs or modifications that apply to these areas.*

d. *Similar to Item 3 above, VFDR dispositions state that risk, DID, and safety margin criteria are met without further action [emphasis added] and then list modifications or DID RAs. Please confirm that the modifications and RAs listed in the VFDR dispositions are necessary to satisfy the risk, DID, or safety margin criteria.*

<u>Waterford 3 Response</u>

The information/results for Fire Areas RAB 1, RAB 5 and RAB 6 have been updated as a result of the analysis work performed to support the LAR Supplement. Cited VFDRs are no longer associated with Recovery Actions. Attachments C, G, S and W were revised accordingly in the LAR Supplement (See W3F1-2013-0048).

Safe Shutdown RAI 14 (RAI SSA 14)

The disposition statement in LAR Attachment C, VFDR 8C-30 is incomplete. Please provide the corrected statement.

Waterford 3 Response

VFDR 8C-30 no longer applies and was removed from the B-3 Table in the LAR Supplement (See W3F1-2013-0048).

Safe Shutdown RAI 15 (RAI SSA 15)

Under the "Basis" heading in LAR Attachment K, Deviation-42, Item (d) states, "The section of ductwork between the two valves will be provided with a 1-hour fire resistant barrier." This commitment was contained in the original approval request dated September 9, 1983. Please confirm that this 1-hour barrier is installed and remains intact.

Waterford 3 Response

The fire barrier discussed in LAR Attachment K, Deviation 42, Item (d) is installed and remains intact.

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 3 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 3 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 3 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?

NFPA 805 audits and assessments will be conducted under the Waterford 3 Quality Assurance and Fire Protection Programs and procedures. Section C.2 of the Waterford 3 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 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 3 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.

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 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 3 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 B-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 B-1 and update the necessary LAR sections as appropriate to reflect a complete list. For example:

a. *Table B-1,3.8.1 refers to NFPA 72 (1975) but LAR 6.0 does not list NFPA 72*
b. *Table B-1,3.8.1(2) refers to NFPA 720 (1975) but LAR 6.0 does not list NFPA 720*
c. *Table B-1, 3.8.1 (1) refers to NFPA 72E (1974) but LAR 6.0 (6.13) refers to 1975 edition*
d. *Table B-1, 3.3.1.3.1 refers to NFPA 518 but neither Table B-1 nor LAR 6.0 (6.11) contains the edition*
e. *Table B-1, 3.3.1.3.1 refers to NFPA 241 but neither Table B-1 nor LAR 6.0 (6.17) contains the edition*
f. *Table B-1, 3.3.3 refers to NFPA 101 but neither Table B-1 (Reference Document column) nor LAR 6.0 list NFPA 101 nor contain the edition*
g. *Table B-1, 3.3.2 refers to NFPA 220 but neither Table B-1 (Reference Document column) nor LAR 6.0 list NFPA 220 nor contain the edition*
h. *Table B-1,3.3.5.1 refers to IEEE-383 or NFPA 262 but neither Table B-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 B-1 references and their editions and ensure they are captured with Section 6.0 and Table B-1 appropriately and provide justification for not including any B-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.
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.

RAI #	LAR Attachment A (B-1 Table)	LAR 6.0 (References)
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, Sections 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.

Fire Protection Engineering RAI 01.01 (RAI FPE 01.01)

By letter dated September 27, 2012, the license responded to Fire Protection Engineering (FPE) RAI 01. In that response, the licensee committed to Section 3.3.4 of NFPA Standard 220, "Standard on Types of Building Construction," 1999 edition. However, Section 3.3.4 does not exist in the 1999 edition of this standard. Please discuss the apparent error regarding the reference to this section and correct as applicable.

In the same response, the licensee identified a code of record for NFPA Standards 101, "Life Safety Code," and 241, "Standard for Safeguarding Construction, Alteration, and Demolition Operations," that they comply with, but identified complying with only a few selective sections in each standard. However, the licensee may have missed many sections that pertain to the requirements in the NFPA 805 standard. Please explain if it is accurate to state globally, that the licensee is committed to those editions of NFPA 101, 220, and 241, identified in the response to FPE RAI 01, but only to those sections that pertain to the 2001 edition of the NFPA 805 standard. If not, please provide a justification explaining why not.

Waterford 3 Response

Waterford 3's review confirms that the reference to Section 3.3.4 of NFPA Standard 220, 1999 edition is in error. The correct applicable section number of NFPA Standard 220, 1999 edition is Section 2-1 "Noncombustible Material".

Waterford 3 is committed to those sections of NFPA 101, 220, and 241 that pertain to the 2001 edition of the NFPA 805 standard. Waterford 3's review determined the only applicable sections of NFPA 101, 220, and 241 that pertain to the 2001 edition of NFPA 805 are as specified in the following table:

NFPA Code Number	Committed Sections	Applicable NFPA 805 Section
101	10.2.3 and 10.2.7	3.3.3
101	8.2.3.2.1(a) and 9.2.1	3.11.3(3)
220	2-1 Definition of "Noncombustible Material"	3.3.2
241	5.1	3.3.1.3.1

RAI FPE 02

Table B-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.

<u>Waterford 3 Response</u>

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.

<u>RAI FPE 03</u>

Table B-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.

<u>Waterford 3 Response</u>

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 Supplement, Attachment A (Table B-1), Section 3.11.5. (See W3F1-2013-0048)

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.

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 Supplement, Attachment A, Section 3.11.5 identifies that ERFBS's credited for NFPA 805 are only necessary in Fire Area RAB-6. All ERFBS in this fire area credited for NFPA 805 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 3 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.*

<u>Waterford 3 Response</u>

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.

<u>RAI FPE 08</u>

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.

<u>Waterford 3 Response</u>

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 Supplement, Attachment C-2 (See W3F1-2013-0048). 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.*

<u>Waterford 3 Response</u>

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***:

<u>Deviation Regarding Reduced Capacity of the Oil Collection Tanks</u>: 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.

<u>Summary of the Originally Approved Suppression and Detection Configurations</u>: 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.*

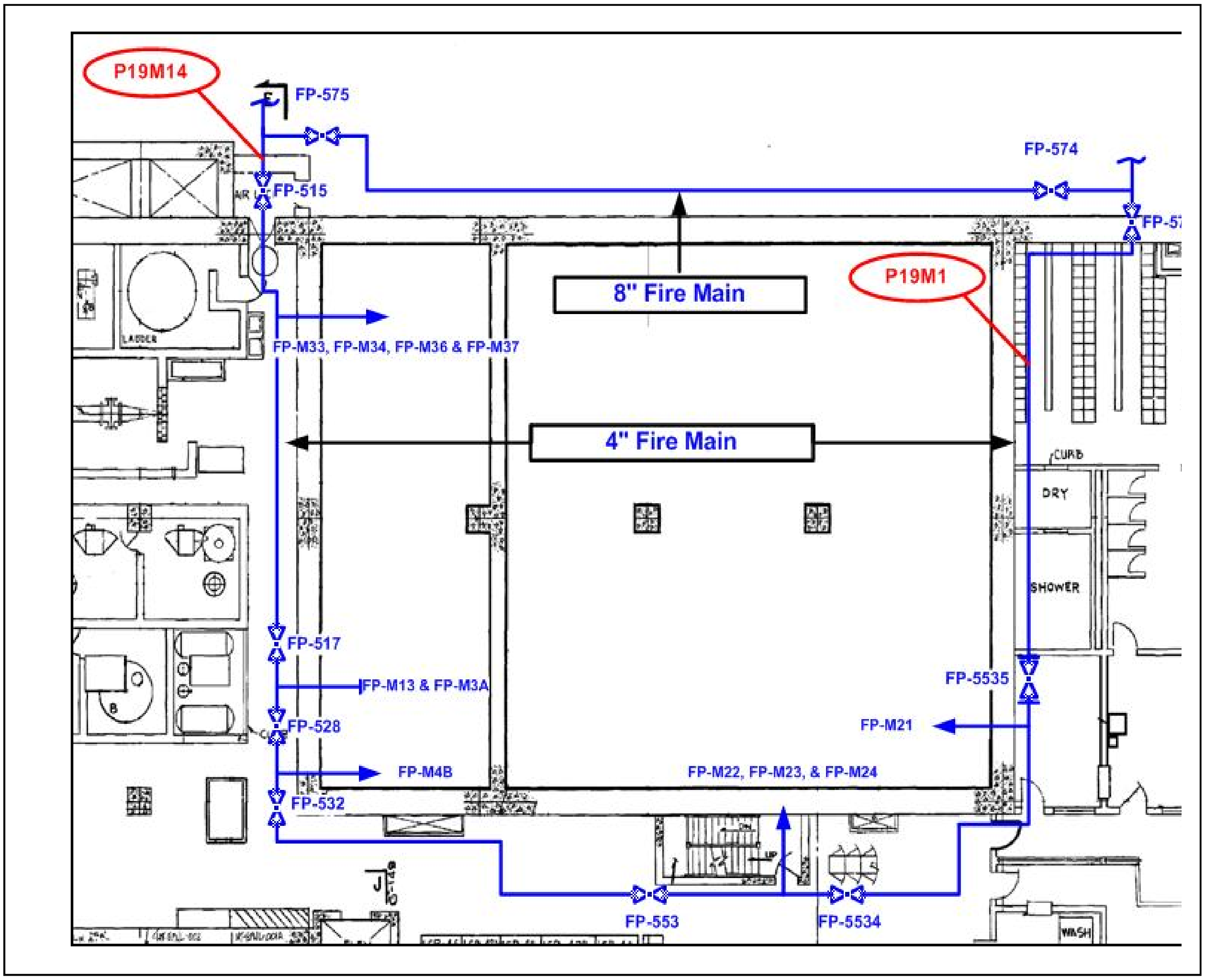
c. *The capacity of the three sprinkler systems given the least demanding portion of the main loop is out of service.*

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. ***Procedures in place to mitigate***:

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. ***Capacity of the three sprinkler systems given the least demanding portion of the main loop is out of service:***

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 placnt 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 3.3.5.3. 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 3.3.5.3.

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.

Fire Protection Engineering RAI 13.01 (RAI FPE 13.01)

By letter dated September 27, 2012, the licensee responded to FPE RAI 13 and made several references to NFPA 805, Section 3.3.5.1. However, the RAI concerns NFPA 805, Section 3.3.5.3. Please clarify the response with regard to the correct NFPA 805 section.

Waterford 3 Response

Noted reference to NFPA 805 Section 3.3.5.3 corrected in response to RAI FPE 13

Fire Protection Engineering RAI 14 (RAI FPE 14)

Table B-3 Suppression System Clarifications

1. *LAR Attachment C (Table B-3), Fire Area RAB 21 (Component Cooling Water Pump B, page C-453), section "Fire Suppression Activities Effect on Nuclear Performance Criteria," states that "This area has a pre-action system." However, LAR Table 4-3, FSAR (Revision 11) page 9.5-60, and the remainder of LAR Table B-3 for Fire Area RAB 21, indicate there is no fire suppression system.*

    a. *Please clarify what type of fixed fire suppression system(s) is installed in Fire Area RAB 21.*

    b. *Please clarify which fixed fire suppression system(s) is credited for NFPA 805 in Fire Area RAB 21.*

2. *LAR Table B-3, Fire Area RAB 37 (Emergency Feedwater Pump A, page C-594) states that "no" suppression installed but contradicts with "a pre-action automatic sprinkler system is provided…". Final Safety Analysis Report (FSAR), Revision 11, page 9.5-60, and LAR Table 4-3 both indicate Fire Area RAB 37 contains a fire suppression system.*

    a. *Please clarify what type of fixed fire suppression system(s) is installed in Fire Area RAB 37.*

b. *Please clarify which fixed fire suppression system(s) is/are credited for NFPA 805 in Fire Area RAB 37.*

3. *LAR Table B-3, Fire Area RCB (Reactor Containment Building, page C-624) states that a "manually actuated water spray system is provided." Under section "Fire Suppression Activities Effect on Nuclear Performance Criteria," it states "no automatic suppression is credited…". This wording is not consistent with other fire areas and appears to indicate there is an automatic system installed.*

   a. *Please clarify what type of fixed fire suppression system(s) is installed in Fire Area RCB.*

   b. *Please clarify which fixed fire suppression system(s) is credited for NFPA 805 in Fire Area RCB.*

<u>Waterford 3 Response</u>

The information/results for Fire Areas RAB 21, RAB 37 and RAB 6 have been updated as a result of the analysis work performed to support the LAR Supplement. Attachment C was revised accordingly in the LAR Supplement (See W3F1-2013-0048).

<u>Fire Protection Engineering RAI 15 (RAI FPE 15)</u>

Table B-3 Suppression Activities' Adverse Effect

In several fire areas, it was not clear whether the adverse effects from all applicable fire suppression activities were considered.

For each fire area within Table B-3, please clarify that the statement "…fire suppression activities will not adversely affect the ability to achieve the nuclear safety performance criteria" accounts for all manual suppression activities including fire brigade, any installed fixed manual suppression system(s), and any installed automatic suppression system(s).

<u>Waterford 3 Response</u>

Only fire areas that contain redundant safe shutdown equipment susceptible to water damage are evaluated for potential adverse effects due to fire suppression activities. An example is fire area RAB 2 where redundant chilled water units are located. The fire suppression effects analysis evaluation performed to substantiate the Table B-3 statement "... fire suppression activities will not adversely affect the ability to achieve the nuclear safety performance criteria" did consider all manual suppression activities including fire brigade use of fire hose streams, installed fixed manual suppression system(s), and installed automatic suppression system(s). This is supported by the following:

1. Electrical cabling at Waterford 3 associated with safe shutdown components is not subject to water damage because the cables were purchased and installed for use in both dry and wet locations. Electrical equipment excluding cables (motors, cabinets, controls, transmitters, relays, etc.) is considered susceptible to water damage.
2. Cable tray and conduit entrances into equipment are equipped with a water seal(s).
3. Flooding calculations have evaluated that the equipment is adequately protected from area flooding (rising water) due to postulated piping cracks, breaks, and/or postulated

actuation of fire suppression systems (fixed manual/automatic suppression systems, including use of fire hose streams).

4. Fire pre-plans and fire brigade training ensure fire brigade members are aware of and capable of applying judicious use of fire hose streams to limit water damage to redundant safe shutdown equipment in the same area.

   However, fire pre-plans and fire brigade training plans need to be enhanced for areas containing redundant safe shutdown equipment susceptible to water damage to ensure fire brigade members are aware of and capable of applying judicious use of fire hose streams to limit water damage to redundant safe shutdown equipment in the same area. LAR Attachment S, Item S2-5 has been revised as follows: “Update Pre-Fire Strategies and necessary plant documents to: 1) Include a description of areas for flooding (currently in S2-5). 2) Identify areas containing redundant safe shutdown equipment susceptible to water damage from fire brigade fire suppression activities. 3) Revise fire brigade training plans to address judicious use of fire hose streams to limit water damage to redundant safe shutdown equipment.

RADIOACTIVE RELEASE (RR)

<u>RAI RR 01</u>

Please provide the specific criteria that were used to "screen out" fire Zones listed as such in Attachment E to the License Amendment Request (LAR).

<u>Waterford 3 Response</u>

Fire area/zones were screened out based on no reasonable potential for contaminated materials to be stored in these areas during all plant operating modes, including full power to and including non-power conditions.

This meets the goal to have a radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, Limits.

The screening process considered input from RP personnel and review of the Pre-Fire Strategies.

Based on the review for this RAI, four additional areas associated with pre-fire strategies will be classified as “screened in”. These areas include RAB 22, RAB 23A, TB-001, and TB-002.

Additional details can be found in engineering report RSC 12-23L, “Resolution of W-3 Radioactive Release RAI RR 01 – Screening Criteria for Fire Zones.”

<u>RAI RR 02</u>

Several Fire Areas are indicated where liquid effluents are collected in sumps or floor drains and routed to holdup tanks. For each such area, please provide a qualitative assessment describing the:

a. Capability of sumps and tanks to contain the estimated amount of water to be generated;

b. Specific actions/methods (e.g., temporary dikes, absorbent materials, directed fire hose spray) needed to ensure containment of the liquid effluents from this area;

c. Additionally, please discuss any pre-planned mitigation actions, procedures, and training.

<u>Waterford 3 Response</u>

a. Report RSC 12-24L identifies the W-3 fire areas where liquid effluents are collected in sumps or floor drains and are routed to holdup tanks, including the specific drain system associated with each fire area, and any specific actions actions/methods needed to ensure containment of the liquid effluents. This report concludes that the capability of the drain systems is adequate to contain the firefighting liquid effluents.

b. Specific actions to contain fire water introduced as a result of fire fighting efforts are provided in the applicable pre-fire strategy. Because there are drains associated with each area that route the potentially contaminated liquid effluents to a sump or drain system that has the capability to treat contaminated water, no specific actions/methods were determined to be required to contain the liquid effluents from the areas of interest with the exception of fire area RSB, “Radwaste Solidification Building.” For a fire in this area, doors need to be opened to allow the liquid effluent into the LWM system drains.

c. NTP-202 “Fire Brigade Training Program” provides the fire brigade training sequence to assure the capability to fight potential fires is established and maintained. More details that provide the pre-planned mitigation actions, procedures, and training associated with fighting fires in radiological areas, including the handling of contaminated runoff from the fire are contained in report RSC 12-24L.

RAI RR 03

Please clarify the method used for "manual ventilation," and provide a qualitative or quantitative assessment of the gaseous releases from such methods for each applicable area that will demonstrate that it meets the acceptance criteria for the Instantaneous Release Technical Specification. Also, for areas where normal ventilation may not be available, include a description of the:

a. *Type of fire most likely to occur in that fire area (e.g., electrical, transient combustibles);*
b. *Type and amount of radioactive contamination in the fire area;*
c. *Type of fire suppression used in the area (e.g., water, foam, Halon, C02);*
d. *Duration of anticipated fire fighting activities;*
e. *Actions/methods needed to minimize and/or monitor the release of the contaminated gaseous effluent;*
f. *Describe how the Technical Specification limit will be met.*

Waterford 3 Response

Manual ventilation is the use of smoke ejectors to remove smoke from an area after a fire has occurred and normal ventilation is either unavailable or inadequate to remove the smoke. Smoke yield for fires in the RAB and/or FHB areas are not expected to be greater than the capacity of the HEPA filter. Because of little or no contamination in these fire areas, HEPA filters can be bypassed without any impact to unrestricted release of smoke. For areas other than the RAB and FHB, a dispersion analysis was performed and documented in Technical Support Document 12-085 that shows that the gaseous effluent releases from the bounding radiological release fire(s) at WSES-3 meet the Technical Specification acceptance criteria. RSC-CALKNX-2012-0904 identifies the fire areas that have the potential to use smoke ejectors as a backup to normal ventilation, the fire areas that do not have any installed ventilation system in them, and the actions/methods needed to minimize and/or monitor the gaseous effluent releases.

- Questions a, b, c & d: RSC-CALKNX-2012-0904 Table 2 identifies the type(s) of fire most likely to occur in each fire area, the type and amount of radioactive contamination in each area, the type(s) of fire suppression used in each area, and the duration of anticipated fire fighting activities in each area.

- Question e: Actions/methods needed to minimize and/or monitor the release of the contaminated gaseous effluent include:
    - Use of portable smoke ejectors, including air sampling methods when they are in use, and specifying that smoke removal via a pathway that contains a functioning ventilation system is the preferred pathway
    - Air sampling in a smoke-filled environment when smoke ejectors are not in use.

Radiation Protection procedures will incorporate the above actions.

Question f: Bounding scenarios for each waste stream are documented in Technical Support Document 12-085. The results for the most limiting source terms which lead to the highest fraction of the dose limit for gaseous effluents is the LWM Resin in the RSB which results in a release that is 11.6% of the technical specification limits.

RAI RR 04

Please describe (for areas where drains and/or sumps are not provided) engineered provisions to monitor and contain liquid fire fighting effluent and provide a bounding, quantitative or qualitative analysis that identifies the maximum quantities, forms of radioactive materials in the fire areas, estimates of the effluent concentrations discharged to the unrestricted area, and demonstrates that the instantaneous dose rate limit of the Technical Specifications would be met. Please describe specific methods in the fire pre-plans that will be used to limit or prevent these liquid releases to the unrestricted area (e.g., spill control kits, temporary dikes, storm drain covers, settling ponds etc.).

<u>Waterford 3 Response</u>

a. RSC-CALKNX-2012-0905 documents five fire areas that do not have drains or sumps associated with them. Engineered provisions such as dikes and the Arpent Canal exist to contain liquid fire fighting effluents. Procedural controls exist to monitor any release from the canal prior to discharge.

b. RSC-CALKNX-2012-0905 provides the information on the fixed contamination levels present in the areas, and the types and maximum quantities of radiological materials that can be stored in each area. Bounding scenarios for each waste stream are documented in Technical Support Document 12-085. The results for the most limiting source terms which lead to the highest fraction of the dose limit for liquid effluents is the DAW stored in the LLWRS which results in a release that is 0.377% of the technical specification instantaneous dose rate limits.

c. The Arpent Canal currently functions similar to a settling pond. Pre-fire strategies will be updated to include additional provisions to limit or prevent a liquid release to the Arpent Canal.

RAI RR 05

Attachment E states that there are sealed containers of low level radioactive material as well as fixed contamination located in the Yard and Outlying Buildings. Please provide the following details:

a. *Type of fire most likely to occur in that fire area (e.g., electrical, transient combustibles);*

b. *Type and amount of radioactive contamination in the fire area;*

c. *Type of fire suppression used in the area (e.g., water, foam, Halon, C02);*

d. *Duration of anticipated fire fighting activities;*

e. *Actions/methods needed to minimize and/or monitor the release of the contaminated gaseous effluent;*

f. *Please describe how the Technical Specification limit will be met.*

Waterford 3 Response

The fire areas associated with the Yard and Outlying Buildings include the LLRWSF, CP, RMSB, RSB, YD, and OCA. A dispersion analysis was performed and documented in Technical Support Document 12-085 that shows that the gaseous effluent releases from the bounding radiological release fire(s) at WSES-3 meet the Technical Specification acceptance criteria. RSC-CALKNX-2012-0904 identifies the specifics associated with each of these fire areas.

- Response to questions a, b, c & d include: Table 2 of RSC-CALKNX-2012-0904 Identifies the type(s) of fire most likely to occur in each fire area, the type and amount of radioactive contamination in each area, the type(s) of fire suppression used in each area, and the duration of anticipated fire fighting activities in each area.
- Response to question e: Actions/methods needed to minimize and/or monitor the release of the contaminated gaseous effluent includes air sampling for smoke filled facilities when necessary.

Radiation Protection procedures will incorporate the above actions.

Question f: Bounding scenarios for each waste stream are documented in Technical Support Document 12-085. The results for the most limiting source terms which lead to the highest fraction of the dose limit for gaseous effluents is the LWM Resin in the RSB which results in a release that is 11.6% of the technical specification limits.

RAI RR 06

Describe if all modes of operations, including non-power operations, have been considered.

Attachment E states that there are sealed containers of low level radioactive material as well as fixed contamination located in the Yard and Outlying Buildings. Describe what isotopes will be present and what the activity is.

Waterford 3 Response

All modes of operation including non-power operations have been considered.

A table is provided in calculation RSC-CALKNX-2012-0906 containing information on the isotopes present in each low level waste stream stored in the sealed containers including the activities associated with each isotope. The characterization data and isotopic information for the DAW, filters, plant resins and LWM resins are taken from the 2011 10 CFR Part 61 report on waste streams at WSES-3. The characterization data and isotopic information for the powdex and B/D resins are taken from the LLRWSF Shielding Calculation. Contaminated oil characterization data and isotopic information are taken from the shipping data associated with the last shipment of contaminated oil from the site.

The radiological surveys used to determine the highest fixed contamination level in each area are contained in Attachment A of calculation RSC-CALKNX-2012-0906.