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ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Duke Energy Carolinas, LLC (Duke Energy)
McGuire Nuclear Station (MNS), Units 1 and 2
Docket Numbers 50-369, 50-370
Renewed License Numbers NPF-9 and NPF-17

Subject: Response to August 28, 2014 NRC Request for Additional Information Regarding License Amendment Request To Implement A Risk-Informed Performance-Based Fire Protection Program (TAC NOs. MF2934 and MF2935).

References:

1. MNS Letter, License Amendment Request (LAR) to Adopt National Fire Protection Association (NFPA) 805 Performance-Based Standard for Fire Protection for Light-Water Reactor Generating Plants, dated September 26, 2013, Agencywide Document and Management System (ADAMS) Accession Number ML13276A126.
2. NRC E-Mail, McGuire 1 and 2 NFPA 805 License Amendment Request - Unacceptable With The Opportunity To Supplement, dated December 18, 2013, ADAMS Accession Number ML13352A514.
3. MNS Letter, Supplemental Information For License Amendment Request (LAR) to Adopt National Fire Protection Association (NFPA) 805 Performance-Based Standard for Fire Protection for Light-Water Reactor Generating Plants, dated January 8, 2014, No ADAMS Number.
4. NRC Letter, McGuire Nuclear Station, Units 1 and 2 - Acceptance of Requested Licensing Action RE: License Amendment Request to Adopt National Fire Protection Association (NFPA) 805 Performance-Based Standard for Fire Protection for Light-Water Reactor Generating Plants (TAC NOs. MF2934 and MF2935), dated January 15, 2014, ADAMS Accession Number ML14014A279).
5. NRC Letter, Request for Information Regarding License Amendment Request To Implement A Risk-Informed Performance-Based Fire Protection Program (TAC NOs. MF2934 and MF2935), dated August 28, 2014, ADAMS Accession Number ML14233A366).

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By letter dated September 26, 2013 (Reference 1), Duke Energy submitted a license amendment request (LAR) to adopt a new, risk-informed, performance-based (RI-PB) fire protection licensing basis for the MNS Unit Nos. 1 and 2.

On December 18, 2013 (Reference 2), the NRC requested supplemental information in order to make the September 26, 2013, LAR complete and acceptable for review by the NRC. By letter dated January 8, 2014 (Reference 3), Duke Energy provided the requested supplemental information to the NRC. By letter dated January 15, 2014 (Reference 4), the NRC accepted the September 26, 2013, LAR for review.

By letter dated August 28, 2014 (Reference 5), the NRC requested additional information (RAI) in order to complete their review of the September 26, 2013, LAR. That letter grouped the RAIs into 60-day, 90-day, and 120-day response times. Duke Energy's response to the 60-day RAIs are provided in Enclosure 1. As a result of some of the responses in Enclosure 1, it will be necessary to revise some pages of the LAR. Those LAR revisions, which are described in the applicable response, have been entered into the MNS Corrective Action Program and will be included in the submittal providing the responses to the 120-day RAIs. Responses for the 90-day and 120-day RAIs will be provided by November 12, 2014, and December 12, 2014, respectively.

The conclusions reached in the original determination that the September 26, 2013, LAR contains No Significant Hazards Considerations and the categorical exclusion from performing an Environmental/Impact Statement have not changed as a result of the August 28, 2014, RAIs and the RAI responses in Enclosure 1.

This submittal does not contain any new or revised regulatory commitments.

Please direct any questions on this matter to Jeffrey N. Robertson at 980-875-4499.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 13, 2014.

Sincerely,

Steven D. Capps

Enclosure 1

xc:

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ENCLOSURE 1

Duke Energy Responses To The August 28, 2014 60-Day RAIs Related To The MNS NFPA 805 LAR

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO ADOPT

NATIONAL FIRE PROTECTION ASSOCIATION STANDARD 805

PERFORMANCE BASED STANDARD FOR FIRE PROTECTION

FOR LIGHT WATER REACTOR GENERATING PLANTS

DUKE ENERGY CAROLINAS, LLC

MCGUIRE NUCLEAR STATION UNITS 1 AND 2

DOCKET NOS. 50-369, 50-370

By letter dated September 26, 2013, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13276A126), Duke Energy Carolinas (Duke) submitted a license amendment request to change its fire protection program to one based on the National Fire Protection Association (NFPA) Standard-805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition, as incorporated into Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.48(c). In order for the U. S. Nuclear Regulatory Commission (NRC) staff to complete its review of the license amendment request (LAR), the following additional information is requested:

Programmatic Requests For Additional Information (RAIs) - 60-Day Responses

Programmatic RAI 01

Based on the NRC Staff's review of the LAR and associated documentation, it was determined that the LAR did not provide the information needed for the NRC staff to evaluate what changes will be made to the site Quality Assurance (QA) program as well as to site procedures to incorporate NFPA 805 requirements.

Describe the changes to the QA program and site procedures to ensure NFPA 805 fire protection requirements are incorporated into existing processes and programs. Further, discuss how NFPA 805 Section 2.7.3 requirements are or will be included within and implemented by the existing QA program and any planned modifications.

Duke Energy Response:

The Duke Energy Carolinas (DEC) QA Program Topical Report was revised in Revision 40 to update the definition of QA 3 as follows:

> "QA Condition 3 covers those systems, components, items, and services which are important to fire protection in addressing 10 CFR 50.48."

Revising the QA Topical Report to address 10 CFR 50.48 provides application to pre-transitional license basis, as well as NFPA 805 license basis criteria and expands the scope of the fire protection QA program to include those elements required by NFPA 805 Chapter 4 and NFPA 805 Section 2.7.3. As part of the NFPA 805 implementation Engineering Change (EC) applicable site procedures will be revised to ensure NFPA 805 fire protection requirements are incorporated into existing processes and programs. Implementation items 1, 3, 4, 5, 6, 7, 8, and 18 are examples.

Fire Protection Engineering (FPE) RAIs - 60-Day Responses

FPE RAI 01

LAR (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13276A126) Attachment A, Section 3.4.1(c) states that fire brigade members are plant operators and "qualifications of individuals in the fire protection organization are administratively controlled to ensure qualification of the individual commensurate with the position being held and activities being performed." NFPA Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants" (NFPA-805), Section 3.4.1(c) requires that the fire brigade leader and at least two brigade members have sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppressants on nuclear safety performance criteria. In Section 1.6.4.1, "Qualifications" of NRC Regulatory Guide (RG) 1.189, "Fire Protection for Nuclear Power Plants", Revision 2, September 2009, (ADAMS Accession No. ML092580550) the NRC staff has acknowledged the following example for the fire brigade leader as sufficient:

> The brigade leader should be competent to assess the potential safety consequences of a fire and advise control room personnel. Such competence by the brigade leader may be evidenced by possession of an operator's license or equivalent knowledge of plant systems.

Provide additional detail regarding the training provided to the fire brigade leader and members that addresses their ability to assess the effects of fire and fire suppressants on nuclear safety performance criteria.

Duke Energy Response:

McGuire Nuclear Station (MNS) utilizes a fire brigade where each shift has a brigade leader and at least two brigade members with sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppressants on nuclear safety performance. This is consistent with NFPA 805 Chapter 3 (Section 3.4.1(c)) and Duke Energy procedures NSD 112, Fire Brigade Organization, Training and Responsibilities and AD-EG-ALL-1530, NFPA 805 Fire Brigade Training requirements. NSD 112 is to be superseded by AD-EG-ALL-1530.

An equivalent knowledge of plant systems is provided for in NSD 112, Fire Brigade Organization, Training and Responsibilities and AD-EG-ALL-1530, NFPA 805 Fire Brigade Training. These procedures specify that training and knowledge of the effects of fire and fire suppressants on safe shutdown equipment may be satisfied by completion of reactor operator certification training or training in specifically identified plant systems.

FPE RAI 02

LAR Attachment A, Section 3.11.5 states that Electrical Raceway Fire Barrier Systems (ERFBS) such as Thermo-Lag, 3M Interam, Hemyc, MT, or Darmatt systems are not utilized for Chapter 4 compliance. However, in LAR Attachment C, Table B-3, Hemyc is cited by engineering evaluations as adequate for the hazard in fire areas 2A (Unit 1 Turbine Driven CA Pump Room), and 3A (Unit 2 Turbine Driven CA Pump Room).

Provide clarification of the use of Hemyc materials. If Hemyc is used in a NFPA-805 compliance basis, then provide a description and location of the credited Hemyc fire barriers used for the Nuclear Safety Capabilities Assessment (NSCA). Provide the basis for credited ratings of the

barriers as ERFBS. Identify and briefly describe any proposed plant modifications to barriers using Hemyc or MT. Identify if any compensatory measures are currently in place and the justification for their use, and whether compensatory measures will remain after completion of any proposed plant modifications. If performance-based methods are used, include consideration of safety margin and defense-in-depth (DID) in the evaluations.

Duke Energy Response:

Hemyc is not used as a MNS NFPA 805 compliance basis. Hemyc is not required at MNS as part of the NFPA 805 analysis. The Variations From Deterministic Requirement's (VFDRs) and the engineering evaluation that discuss Hemyc are identified in LAR Attachment C, Table C-1 – NFPA 805 Ch 4 Compliance (NEI 04-02 Table B-3) in Fire Areas 2A and 3A. These VFDRs are specifically analyzed without any credit for Hemyc. As stated in the disposition to the VFDRs, risk, defense in depth, and safety margin are satisfied without further action. Therefore, no modifications are proposed for any of the Hemyc installations. The inclusion of the engineering evaluation crediting Hemyc was an oversight. The Hemyc engineering evaluation should not have been identified in Table C-1 and identified as required for transition.

The engineering evaluation crediting Hemyc (MCC-1435.03-00-0014, Part 1) in Attachment C, Table C-1, Fire Areas 2A and 3A will be deleted. A revision to LAR Attachment C, Table C-1 for this item is planned to be submitted with the 120 day RAI responses.

FPE RAI 04

LAR Attachment L, Approval Request 2 requests to provide a performance-based evaluation in place of the NFPA-805 Section 3.3.5.1 requirement that wiring above suspended ceiling shall be kept to a minimum and where installed, electrical wiring shall be listed for plenum use, routed in armored cable, routed in metallic conduit, or routed in cable trays with solid metal top and bottom covers. The LAR stated that the wiring/cable is installed above suspended ceilings which may not comply with the requirements and that the wiring has no impact to nuclear safety.

Describe the proximity of these unqualified cables to nuclear safety capability components or cables, and address the likelihood and significance of potential fires adjacent to those nuclear safety capability components or cables. Additionally, describe what mechanisms are in place to prevent future non-code compliant installations.

Duke Energy Response:

The Control Room is the only location in Power Block that has cables above suspended ceilings that are located near nuclear safety capability equipment (components or cables). Minimum amount of cables exist above Control Room ceiling. These cables are of two types:

- Electrical cabling
- Low voltage communication, video, and data cables

All electrical cables used for power, control, and instrumentation is armored cable or routed in metallic conduit. Cables used for power, control, and instrumentation meet the requirements of NFPA 805 Section 3.3.5.1.

Use of low voltage communication, video, and data cables is minimized. These cables may not be plenum rated which is why LAR Attachment L, Approval Request 2 was submitted. The closest point these unqualified cables are to any nuclear safety capability equipment is

approximately 2 feet, but the majority of the cabling is located more than 7 feet from any nuclear safety capability equipment.

The basis for approval, as documented in Approval Request 2, encompasses the control room in addition to offices and corridors. These low voltage cables are not generally susceptible to shorts, which would result in a fire. In addition, impact of a fire involving these small cables on a target 2 feet away is highly unlikely.

In the unlikely event a fire were to occur, the Control Room is continuously occupied and provided with a smoke detection system compliant with NFPA 72. Should a fire occur, it would be detected and extinguished in the incipient stages before a significant fire could develop.

Implementation Item 7 in Table S-3 of Attachment S addresses control to prevent future non-compliance. LAR Attachment A, Section 3.3.5.1 cites Implementation Item 7. Approval Request 2 also cites Implementation Item 7 as follows:

> *See Implementation Item 7 in Table S-3 of Attachment S for an implementation item to change electrical installation criteria to ensure all future cable installation above suspended ceilings meets the requirements of NFPA 805 Section 3.3.5.1.*

LAR Attachment S, Table S-3, Implementation Item 7 references Attachment A, Section 3.3.5.1 but does not reference Approval Request 2. Table S-3 will be updated to reference LAR Attachment L, Approval Request 2.

A revision to LAR Attachment S, Table S-3, for this item is planned to be submitted with the 120 day RAI responses.

FPE RAI 05

LAR Attachment L, Approval Request 6 provides for a performance-based evaluation for the non-dedicated use of the fire water system.

For those evolutions that initiate and control non-dedicated fire water use of the fire protection system:

a) Describe the administrative controls in place for these evolutions to occur;
b) Describe how approvals are obtained to establish these alternative uses;
c) Describe whether these activities are conducted simultaneously; and
d) Describe how they are controlled in the event of a fire.

Duke Energy Response:

There are three cases for non-dedicated fire water use at MNS, as cited in LAR Attachment L, Approval Request 6. These cases are:

- Back-up to the condenser circulating water pump seal/bearings – automatic alignment, ~200 gpm
- Back-up to the low level intake pump seal/bearings – manual alignment, ~30 gpm
- Intake screen backwash – manual alignment, ~1,100 gpm

The three fire pumps are each rated for 2,500 gpm at 327 head feet (~142 psi). Each fire pump is required to provide 2,500 gpm at 125 psi per Selected Licensee Commitment (SLC) 16.9.1 based on testing every 18 months. The most demanding suppression system is 2,583 gpm at 72 psi (Unit 2 Turbine Building Mezzanine Area 2 (South)). The MNS fire pump curves associated with the

recent fire pump test demonstrate that approximately 4,500 gpm is available at a pressure of 72 psi. The table below illustrates the total demand on the Fire Protection System (RF/RY) by these nondedicated/alternative uses.

Table 1 – Total Fire Water System Demand

Non-Fire Water Use	**Alignment**	**Most Demanding Suppression System Demand at 72 psi**	**Hose Stream (NFPA 805 Section 3.5.1)**	**Non-Fire Water Use Demand**	**Total Demand**
Back-up to the condenser circulating water pump seal/bearings	Automatic	2,583 gpm	500 gpm	200 gpm	3,283 gpm
Back-up to the low level intake pump seal/bearings	Manual	2,583 gpm	500 gpm	30 gpm	3,113 gpm
Intake screen backwash	Manual	2,583 gpm	500 gpm	1,100 gpm	4,183 gpm

RF/RY has sufficient supply for the largest suppression system, hose stream, and the condenser circulating water pump seal/bearings demand as well as the intake pump seal/bearings demand. These demands are well within the capabilities of the fire water system.

RF/RY also has sufficient supply for the largest suppression system, hose stream, and the intake screen backwash operation. When the fire pump is used for backwash operations, it is declared inoperable per procedure OP/0/B/6400/001 C. Therefore, the operators would be aware of on-going operations, and in the event of a fire, the control room could secure intake screen backwash operations to ensure adequate fire protection water is available given the overall large simultaneous demand.

There is the potential that a fire could occur during the alternative use thereby resulting in simultaneous activity. Since back-up to the condenser circulating water pump seal/bearings is automatic, there is the potential that the demand could initiate during one of the manual operations. Simultaneous operation of the back-up to the condenser circulating water pump seal/bearings and the back-up to the low level intake pump seal/bearings would not adversely affect the ability of RF/RY to supply fire water demands. If the intake screen backwash was being performed, then the backwash activity can be secured as needed to ensure proper fire protection water is available.

Administrative Controls and Approvals:

Back-up to the condenser circulating water pump seal/bearings – Alignment of RF/RY to the condenser circulating water pump seal/bearings is performed automatically. There are no administrative controls or approvals for this alignment to occur. Back-up to the condenser circulating water pump seal/bearings demand is approximately 200 gpm. 200 gpm is well within the capacity of RF/RY with the largest suppression system and hose stream flowing.

Back-up to the low level intake pump seal/bearings – Alignment of RF/RY for back-up cooling to the low level intake pump bearings and seal is performed by implementation of procedure OP/1/B/6400/001 F, Low Level Intake System. Explicit approval for the use of RF/RY for backup is not identified. In order for this alignment to occur, there must be a failure of other plant equipment and then implementation of a station procedure. It is noted that using RF/RY is a backup if the Low Level Seal Injection Pump is not available. The low level intake pump seal/bearings demand is approximately 30 gpm. 30 gpm is well within the capacity of RF/RY with the largest suppression system and hose stream flowing.

Intake screen backwash – Alignment and approval for the use of RF/RY for intake screen backwash is performed by implementation of procedure OP/0/B/6400/001 C, Intake Screen Backwash System. It is noted that using RF/RY is a backup if the Intake Screen Backwash (RS) Pump is not available. The procedure states that the main fire pump must not be in use to supply RF/RY header pressure as part of the procedure initial conditions. In addition, the fire pump in use for backwash is declared inoperable, thereby identifying to the control room fire pump/water system operation.

A revision to LAR Attachment A and L for this item is planned to be submitted with the 120 day RAI responses that will identify the largest suppression system demand is 2,583 gpm.

FPE RAI 06

LAR Attachment K, Licensing Action 11, identifies that the licensing action will transition. However, the licensee stated that MCC-1435.00-00-0033, "NFPA 20 Code Compliant Report" evaluated fire pumps A, B, and C present compliance with NFPA 20, "Standard for the Installation of Stationary Pumps for Fire Protection," 2007 edition. The licensee further stated that once the non-compliances identified are addressed, the fire pumps will be considered functionally equivalent and therefore, the licensing action is not required for transition.

Provide clarification with regard to the desired transition compliance method. Additionally, lack of compliance with NFPA 20 is identified in LAR Attachment L, Approval Request 5. Describe the differences between Licensing Action 11 and Approval Request 5.

Duke Energy Response:

The last paragraph of LAR Attachment K, Licensing Action 11, Licensing Basis section is incorrect. This licensing action is required post transition.

The last two sentences of LAR Attachment K, Licensing Action 11, Licensing Basis section, state "MCC-1435.00-00-0033, NFPA 20 Code Compliant Report, evaluated the MNS fire pumps A, B, and C present compliance with NFPA 20, 2007 edition. Once the non-compliances identified are addressed, the fire pumps will be considered functionally equivalent. Licensing action is, therefore, not required for transition." This statement will be will be deleted and revised to state " The bases for previous acceptance remain valid."

The Licensing Action and the Approval Request for NFPA 20 address different sections of NFPA 20. There is no relation between Licensing Action 11 and Approval Request 5. The focus of Licensing Action 11 is the location of the fire pump controllers, specific fire pump/controller design features, and that the pumps/controllers are not UL Listed/FM Approved. The focus of the Approval Request 5 is ability to remotely stop Fire Pumps A and B. Specifically the NFPA 20 sections are outlined as follows:

- Licensing Action 11 cites NFPA 20 Sections 7-2.1, 5-2.1, 7-5.2.5, 7-1.1.1, and 2-2.
- Approval Request 5 cites NFPA 20 Section 7-5.2.3.

A revision to LAR Attachment K for this item is planned to be submitted with the 120 day RAI responses that will include this change.

FPE RAI 07

LAR Attachment C, Table C-2 provides a consolidated summary of the required fire protection systems and features as identified for each fire area. In general, the licensee has identified where required systems are installed on a room basis, however, in the Turbine Building (TB), for example, the licensee has only identified that required suppression and detection is installed. The staff could not determine if all systems in the Turbine Building are required or only certain systems. In addition, LAR Attachment C, Table C-2 identifies in TB1 and TB2 "Water Suppression" and "Suppression" and "Detection System" and "Detection" as required but does not specify the type of suppression or detection system (e.g., sprinkler, deluge, or preaction; smoke or UV).

Provide clarification regarding the types of suppression and detection systems provided in the TB and include a description of which systems are required.

Duke Energy Response:

The TB suppression and detection systems include general area automatic sprinkler systems, automatic water spray systems for hazard protection, ionization smoke detection, flame detection, multi-sensor detection, and heat detection for actuation of the water spray systems.

Required TB suppression and detection systems are identified in the MNS Fire Risk Evaluation (FRE) calculation. These systems were not explicitly identified in LAR Attachment C, Table C-2 – NFPA 805 Required Fire Protection Systems and Features as Table C-2 was generated as a summary level table.

The Fire Area TB1 and Fire Area TB2 FRE DID evaluation identifies that installed detection and suppression systems are required to meet DID criteria (Hazard specific only). These hazard specific systems are the dedicated fixed spray suppression systems (and associated detection for actuation) provided near the large oil related fire hazards to support the Fire PRA (FPRA) qualitative assessment to mitigate the potential for structural collapse due to a fire.

The required TB Suppression Systems are as follows:

Unit 1

- Main Feedwater (CF) Pump automatic water spray system (actuated via fixed temperature heat detectors)
- Hydrogen Seal Oil automatic water spray system (actuated via fixed temperature heat detectors)
- Oil Purifier automatic water spray system (actuated via fixed temperature heat detectors)

- Turbine Lube Oil Reservoir (MTOT) automatic water spray system (actuated via fixed temperature heat detectors)
- Turbine Lube Oil Transfer Tank automatic water spray system (actuated via fixed temperature heat detectors)
- Turbine Piping and Bearing automatic water spray system (actuated via fixed temperature heat detectors)

Unit 2

- Main CF Pump automatic water spray system (actuated via fixed temperature heat detectors)
- Hydrogen Seal Oil automatic water spray system (actuated via fixed temperature heat detectors)
- Oil Purifier automatic water spray system (actuated via fixed temperature heat detectors)
- MTOT automatic water spray system (actuated via fixed temperature heat detectors)
- Turbine Lube Oil Transfer Tank automatic water spray system (actuated via fixed temperature heat detectors)
- Turbine Piping and Bearing automatic water spray system (actuated via fixed temperature heat detectors)

A revision to LAR Attachment C, Table C-2 for this item is planned to be submitted with the 120 day RAI responses.

FPE RAI 08

For the existing Appendix R deviations being transitioned as identified in LAR Attachment K, several licensing actions rely on fire protection features which do not appear in LAR Attachment C, Table C-2.

Licensing Action 01, the fire protection feature relied upon in this deviation included silicon dioxide insulated cable as a 3-hour rated barrier, but in LAR Attachment C, Table C-2 for fire area 9-11, the cable is identified as only required for risk.

Provide a clarification for this, and any similar discrepancies, for example, systems relied on for engineering evaluations.

Duke Energy Response:

The silicon dioxide insulated cable (Meggitt Cable) is identified as MI Cable in Fire Area 09-11, Room 705 in LAR Attachment C, Table C-2. The MI Cable will be identified with an "E" in addition to the "R" for risk. The "E" identifies that the feature is required by engineering evaluation or licensing action. The new entry will read as follows: "MI on some Train A cables: E, R".

An additional review of required systems and features was performed. The review determined that in Fire Areas 32 and 33, MI cable is being used as a radiant energy shield. This is cited in Table C-2 as required for DID. There is an engineering evaluation, which evaluates the MI cable for adequate separation; therefore, this MI cable should also be identified with an "E" for engineering evaluation. The new entries for FA 32, Room 1RB-3 and FA 33, Room 2RB-3 will read as follows: "MI as radiant energy shield on cables: D, E".

A revision to LAR Attachment C, Table C-2 for this item is planned to be submitted with the 120 day RAI responses.

FPE RAI 10

NFPA-805, Section 3.3.7.2, requires that "Outdoor high-pressure flammable gas storage containers shall be located so that the long axis is not pointed at buildings." In LAR Attachment A, the compliance statement for this element is listed as "Complies via Use of EEEE," with a reference to the licensee calculation that should demonstrate compliance of the installed configuration. However, the staff noted that the referenced licensee document does not make a conclusion regarding acceptability. The staff also noted that the text in the compliance basis does not align with the compliance statement for this element.

Therefore, provide:

a. A summary of the calculation which demonstrates the acceptability of the installed configuration, including the key assumptions, results, and acceptance criteria;
b. A revised compliance statement for this NFPA-805 Chapter 3 element which references the correct document; and
c. A revised compliance basis which aligns with the compliance statement.

Duke Energy Response:

The incorrect calculation was referenced in MNS LAR, Attachment A, Section 3.3.7.2. The correct calculation is MCC-1139.01-00-0040, "Missile Barrier - Air Exhaust, Diesel Generator Building." MCC-1139.01-00-0040 uses the previously cited calculation (MCC-1513.03-00-0001) as input to evaluate a hydrogen storage tank missile impact on the diesel generator walls and the potential to damage safety-related equipment. MCC-1139.01-00-0040 concludes that the (diesel generator) wall is thick enough to prevent penetration, perforation, and local failure.

Attachment A, Section 3.3.7.2 References will be revised. Calculation MCC-1139.01-00-0040 will be listed instead of calculation MCC-1513.03-00-0001.

Attachment A, Section 3.3.7.2, Compliance Basis is incorrect by stating, "The bulk hydrogen storage cylinders are orientated with the long axis parallel to the plant." Parallel orientation of the hydrogen cylinders to the plant will be changed to perpendicular orientation. The revised compliance basis will state "The bulk hydrogen storage cylinders are orientated with the long axis perpendicular to the plant. An evaluation conducted by MNS found this configuration to be acceptable."

A revision to LAR Attachment A for this item is planned to be submitted with the 120 day RAI responses that will include this clarification.

Safe Shutdown Analysis (SSA) RAIs - 60-Day Responses

SSA RAI 02

LAR Attachment B, Table B-2, Section 3.1 [C, Spurious Operation], states that the high/low pressure interfaces consist solely of interface with the residual heat removal (RHR) system in accordance with Safety Evaluation Report, NUREG-0422, "Transient Analysis of the Research Reactor MARIA MC Fuel Elements Using RELAP5 Mod 3.3," Supplement 6. The NRC staff noted that the cited reference, however, does not explicitly state that the RHR is the only high/low pressure interface of concern. NEI 00-01 defines high/low pressure interface as "a subset of components considered for spurious actuation involves reactor coolant pressure boundary (RCPB) components whose spurious operation can lead to an unacceptable loss of reactor pressure vessel/reactor coolant system (RPV/RCS) inventory via an interfacing system loss of coolant accident...selected RCPB valves are defined as high/low pressure interface valve components requiring special consideration and criteria," as endorsed by the NRC through Frequently Asked Question (FAQ) 06-0006, "High-Low Pressure Interface Definition and NEI 00-01/NFPA 805 Discrepancies."

a. Provide more detail with regard to the statement that RHR is the only high/low pressure interface to be evaluated as such. If the basis of this limitation is prior approval by the NRC, then justify why the alignment statement is not "Does Not Align but has Previous Approval," or change the entry to revise the alignment statement.

b. For other reactor coolant boundary valves (e.g., RCS high point vents, RCS letdown isolation valves) that are typically considered high/low pressure interface valves, provide a description of the spurious operation analysis performed for those that justifies not evaluating them as high/low pressure interfaces.

Duke Energy Response:

a. Spurious operations are considered in both the selection of nuclear safety performance criteria functions and systems as well as the cabling associated with the components relied upon to achieve those functions. The NFPA Transition Expert Panel Report for Addressing Potential McGuire Multiple Spurious Operations provides the results of the analysis of multiple spurious operations (MSOs) that were then analyzed in the Safe Shutdown Analysis and FPRA. High/low pressure interfaces are limited to meeting the latest guidance in NEI 00-01, Revision 2. NEI 00-01 Revision 1 states that high/low pressure interfaces result in a LOCA. RG 1.189 Revision 2 Section 5.3.2.c endorses NEI 01-01 Revision 2, which expands the high/low pressure interface definition to a LOCA outside containment. MNS analyzed high/low pressure interfaces resulting in a LOCA outside containment and found that the RHR suction isolation valves are the only ones that meet the definition. Thus, MNS meets the intent of the guidance.

b. High/low pressure interface is defined in NEI 00-01, Revision 2 as a failure that results in a LOCA outside containment. The RHR suction valves from the RCS are the only valves that are in that category. MNS analysis identified that another potential high/low pressure interface is the RHR discharge lines to the RCS, but these were eliminated since they are normally open and there are two check valves in each line upstream of these valves that would have to fail to pressurize the low pressure piping. The letdown and excess letdown valves spurious opening are protected by relief valves to adequately relieve, and these are routed to the Pressurizer Relief Tank (PRT). These are not high/low pressure interfaces,

even though the MSO analysis considered spurious operation of the valves in the letdown and excess letdown lines. Thermal Barrier failure in conjunction with a fire does not need to be considered since it is an independent event.

SSA RAI 03.b

LAR Attachment F, "Fire-Induced Multiple Spurious Operations Resolution," provides a description of the process for evaluating potential multiple spurious operations (MSOs). In order to clarify the methodology, provide the following:

b. LAR Attachment F stated that the expert panel consisted of MNS fire protection and post-fire safe shutdown, McGuire Nuclear Station (MNS) Operations, PRA, and members of the Strategic Alliance for NFPA-805 Transition team. Describe what the "Strategic Alliance" is, and what qualifications or experience they provided the expert panel.

Duke Energy Response:

b. The Strategic Alliance was a group of contractors with expertise relevant to the analysis of MSOs. The Strategic Alliance contractors collectively have experience in safe shutdown analysis, Fire Protection, and electrical systems and circuits. The MSO Expert Panel consisted of individuals in the fire protection, post-fire safe shutdown, plant operations, and FPRA disciplines with a team summary of PRA (two), MNS-SRO previous license (three), other SRO license (two), APP R (six), Instrument and Control (five), Operational procedures (three), and MOV (one).

The MSO calculation, MCC-1435.00-00-0023, contains a brief biography for every member of the Expert Panel, Duke Energy personnel, as well as the Strategic Alliance personnel.

SSA RAI 06

LAR Attachment B, Table B-2, Section 3.2.1.2, "Fire Damage to Mechanical Components," states that heat sensitive piping materials, including tubing with brazed or soldered joints are not included in the assumption of no mechanical damage. The licensee's analysis stated that instrument sensing lines were evaluated as if the fluid boundary remains intact.

Provide the justification for this assumption specifically with regard to heat sensitive piping materials, including tubing with brazed or soldered joints.

Duke Energy Response:

MNS Nuclear Safety Capability Assessment (NSCA) instruments do not use soldered or brazed connections. Process lines, including instrument air lines, use stainless tubing and/or copper tubing. These use compression fittings in accordance with Installation Procedure IP/0/A/3090/25 "Installation Of Instruments, Instrumentation Lines, And Associated Hardware" (which uses parts of MCS-1210.05-00-0038 "Instrumentation and Controls Field Installation Specification" as attachments to the IP). Freon lines on chiller packages/skids could have brazed connections. However, they are considered failed with the chiller package/skid when a fire occurs in their area.

Therefore, the MNS NSCA Aligns with this NEI 00-01 Guidance.

SSA RAI 07

LAR Attachment D describes the methods and results for non-power operations (NPO) transition. Provide the following additional information:

a. A description of any actions that are credited to minimize the impact of fire-induced spurious actuations on power operated valves (e.g., air-operated valves and motor operated valves) during NPO either as pre-fire plant configuring or as required during the fire response recovery.

b. Identify those recovery actions relied upon in NPO and describe how recovery action feasibility is evaluated.

Duke Energy Response:

a. No additional actions beyond normal operating procedures for initial system alignments are credited for NPO. Existing operating procedures, in some cases, require de-energization of key components in the shutdown cooling flowpath to preclude a loss of shutdown cooling event (i.e., the RHR suction valves 1(2)ND VA0001B and 1(2)ND VA0002AC from the RCS), and these could be considered pre-emptive actions. However, this procedure requirement was not used to exclude a pinch point in the analysis. Those locations where a fire-induced spurious operation could impact a Key Safety Function (KSF) are identified, and where complete loss of a KSF occurs, the location is identified as a pinch point for application of additional fire risk management actions during designated Higher Risk Plant Operating States.

b. No recovery actions (RAs) are required to support the NPO analysis assumptions or are used to restore a KSF following a potential fire event during NPO conditions. Evaluation of feasibility is not required.

Fire Modeling (FM) RAIs - 60-Day Responses

FM RAI 01.b

NFPA 805-Section 2.4.3.3 states that the PRA approach, methods, and data shall be acceptable to the NRC. The NRC staff noted that the fire modeling analysis comprised the following:

- The Generic Fire Modeling Treatments (GFMTs) approach was used to determine the Zone of Influence (ZOI) for ignition sources and the time to Hot Gas Layer (HGL) conditions in all fire areas throughout MNS, Unit 1 and 2.
- The Consolidated Fire Growth and Smoke Transport (CFAST) model was used to assess the main control room (MCR) abandonment time calculations.

LAR Section 4.5.1.2, "Fire PRA," states that fire modeling was performed as part of the fire PRA (FPRA) development (NFPA :805 Section 4.2.4.2). Reference is made to Attachment J, "Fire Modeling Verification and Validation," for a discussion of the acceptability of the fire models that were used to develop the FPRA.

Regarding the acceptability of CFAST for the MCR abandonment time calculations:

b. In the MCR abandonment time analysis, it is assumed that the external doors of the MCR open at 15 minutes based on an estimated fire brigade arrival time. Provide a technical justification for the assumption that the fire brigade will arrive 15 minutes after a fire event based on historic drill records or demonstrate that this assumption is conservative.

Duke Energy Response:

b. The FPRA selects the most adverse fire scenario case (regardless of if/when doors are opened) from the MCR abandonment time analysis and incorporates those results into the FPRA.

   The MNS Fire Brigade average time from initial fire alarm signal to fire brigade arrival at the scene for all plant locations is approximately 18 minutes. While the fire brigade does not conduct drills specifically in the MCR, it should be noted that the fire brigade dress-out area is located less than a minute walk away from the MCR. Additionally, there are several factors to consider that cause drill response times to be greater than actual response times during actual fire events:

   - It is industry and MNS practice that fire brigade drills incorporate a degree of training while performing an overall evaluation of the fire brigade, firefighting equipment performance, and plant administrative controls.
   - Fire brigade drills may vary in types of response, speed of response, and use of equipment. A level of proficiency and safety is desired above simply speed of completion during drills.
   - Another factor affecting response times during drills is delays required for compliance with security, administrative controls, radiological controls, and other barriers. During actual fire events, access/egress routes and administrative procedures are expedited for the fire brigade, further decreasing response time.
   - Industry experience also indicates fire brigade response will quicken based on the human behavioral stimuli provided by an actual event.

- Finally, the time required for drill controllers to verbally describe and the fire brigade to visualize the fire conditions adds considerable time to the drill process that would not be present during an actual event.

FM RAI 01.h

NFPA 805-Section 2.4.3.3 states that the PRA approach, methods, and data shall be acceptable to the NRC. The NRC staff noted that the fire modeling analysis comprised the following:

- The Generic Fire Modeling Treatments (GFMTs) approach was used to determine the Zone of Influence (ZOI) for ignition sources and the time to Hot Gas Layer (HGL) conditions in all fire areas throughout MNS, Unit 1 and 2.
- The Consolidated Fire Growth and Smoke Transport (CFAST) model was used to assess the main control room (MCR) abandonment time calculations.

LAR Section 4.5.1.2, "Fire PRA," states that fire modeling was performed as part of the fire PRA (FPRA) development (NFPA :805 Section 4.2.4.2). Reference is made to Attachment J, "Fire Modeling Verification and Validation," for a discussion of the acceptability of the fire models that were used to develop the FPRA.

Specifically regarding the acceptability of the GFMTs approach:

h. The GFMTs approach describes the critical heat flux for a target that if immersed in a thermal plume. Explain how the modification to the critical heat flux was used in the ZOI and HGL timing determinations.

Duke Energy Response:

h. The modified critical heat flux is a means of accounting for both elevated temperatures and flame heat fluxes and was implemented using either a two or a three point (i.e., temperature) treatment in the FPRA. When the modified heat flux is used to establish an ignition source ZOI in an enclosure with an elevated temperature, the ZOI is larger than an ambient temperature based ZOI. Most plant areas use the two point treatment of the modified critical heat flux. The first point corresponds to temperature conditions between ambient and 80°C (176°F) and represents the temperature interval in which the ZOIs such as those documented in the GFMTs are applicable. The second point corresponds to temperature conditions greater than 80°C (176°F) and is conservatively characterized in the FPRA as a full-room burnout. This applies both to targets located in the thermal plume region and to targets that are located outside the thermal plume region.

   The HGL review for ZOI impact utilizes a three point treatment for greater resolution on the risk characterization. The first point corresponds to temperature conditions between ambient and 80°C (176°F) and represents the temperature interval in which the ZOIs for thermoplastic cable targets are applicable. The second point corresponds to temperature conditions greater than 80°C (176°F) but less than 220°C (428°F) and represents the region where the hot gas layer can produce a heat flux up to 5.7 kW/m^2 (0.50 Btu/s^2). The ZOIs for thermoplastic cable targets, which have a heat flux threshold of 5.7 kW/m^2 (0.50 Btu/s), are applicable in this temperature range when used to identify thermoplastic cable targets because the total heat flux at the ZOI boundary is 11.4 kW/m^2 (1.0 Btu/s^2), the generic threshold for thermoset cables per NUREG/CR-6850. The third point corresponds to temperature conditions greater than 220°C (428°F) and is conservatively characterized in the FPRA as a full-room burnout. This applies both to targets located in the thermal plume region and to targets that are located outside the thermal plume region.

FM RAI 01.i

NFPA 805-Section 2.4.3.3 states that the PRA approach, methods, and data shall be acceptable to the NRC. The NRC staff noted that the fire modeling analysis comprised the following:

- The Generic Fire Modeling Treatments (GFMTs) approach was used to determine the Zone of Influence (ZOI) for ignition sources and the time to Hot Gas Layer (HGL) conditions in all fire areas throughout MNS, Unit 1 and 2.
- The Consolidated Fire Growth and Smoke Transport (CFAST) model was used to assess the main control room (MCR) abandonment time calculations.

LAR Section 4.5.1.2, "Fire PRA," states that fire modeling was performed as part of the fire PRA (FPRA) development (NFPA :805 Section 4.2.4.2). Reference is made to Attachment J, "Fire Modeling Verification and Validation," for a discussion of the acceptability of the fire models that were used to develop the FPRA.

Specifically regarding the acceptability of the PRA approach, methods, and data:

i. Identify whether any fire modeling tools and methods have been used in the development of the LAR that are not discussed in LAR Attachment J. One example would be a methodology used to convert damage times for targets in Appendix H of NUREG/CR-6850 to percent damage as a function of heat flux and time.

Duke Energy Response:

i. No other fire modeling tool or method was used outside GFMTs or MCR Abandonment calculation.

FM RAI 02.c

American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) Standard RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications," Part 4, requires damage thresholds be established to support the FPRA. The standard further states that thermal impact(s) must be considered in determining the potential for thermal damage of systems, structures, and components (SSCs) and appropriate temperature and critical heat flux criteria must be used in the analysis.

Provide the following information:

c. The resolution to finding and observation (F&O) FSS-C5-01 refers to a licensee analysis as the basis for concluding that the armored cable, which has a thin PVC exterior jacket or thermoplastic coating, can be treated as a thermoset material and its associated damage criteria. NUREG/CR-6850 Section H.1.3 recommends that the failure criteria for thermoplastic materials should be applied for mixed configurations, unless appropriate justification for treatment as a thermoset material is provided. Provide justification for concluding that the armored cable can be treated as thermoset material. In the response, specifically address whether a thermoplastic fire can form in the immediate vicinity of the cables themselves given that thermoplastic materials melt and can form a burning pool of liquid material, and how the cable tray configurations identified in Section H.1.3 of NUREG/CR-6850 correspond to those at the plant.

Duke Energy Response:

c. From MCC-1535.00-00-0104, Section 6.1, it states that "it is reasonable to expect that cables above electrical cabinets would tend to have the PVC jacket melt and flow away creating voids for the flow of melting materials from other cables. In addition, the ridges that are characteristic of the armor jacketing provide additional free space for the flow of material. As such, it is not expected that pooling of PVC is likely to occur even if multiple layers of armored cables exist." The low likelihood of a burning pool is attributable in large part to MNS not employing solid bottom trays. The heat release rate (HRR) contribution from the small amount of flame-retardant, self-extinguishing jacket material that might collect on the top surface of the ignition source (e.g., electrical panel) is considered negligible in comparison with the peak HRR of the ignition source and is therefore considered insignificant with respect to the postulated target damage.

FM RAI 02.d

American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) Standard RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications," Part 4, requires damage thresholds be established to support the FPRA. The standard further states that thermal impact(s) must be considered in determining the potential for thermal damage of systems, structures, and components (SSCs) and appropriate temperature and critical heat flux criteria must be used in the analysis.

Provide the following information:

d. Describe how cable tray covers, conduits and wraps affect the damage thresholds that were used in the fire modeling analyses.

Duke Energy Response:

d. No FPRA credit was taken for cable tray covers, conduits or wraps.

FM RAI 03

NFPA-805, Section 2.7.3.2, states that each calculational model or numerical method used shall be verified and validated through comparison to test results or comparison to other acceptable models.

LAR Section 4.5.1.2, states that fire modeling was performed as part of the FPRA development (NFPA-805 Section 4.2.4.2). Reference is made to Attachment J, for a discussion of the verification and validation (V&V) of the fire models that were used. Furthermore, LAR Section 4.7.3 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, for any fire modeling tool or method that was used in the development of the LAR or that is identified in the responses to the above fire modeling RAIs, provide the V&V basis if it is not already explicitly provided in the LAR (for example in LAR Attachment J).

Duke Energy Response:

No other fire modeling tool or method was used outside the GFMTs or the MCR Abandonment calculation.

FM RAI 05

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."

LAR Section 4.5.1.2, "Fire PRA," states that fire modeling was performed as part of the FPRA development (NFPA-805 Section 4.2.4.2). This requires that qualified fire modeling and PRA personnel work together. Furthermore, LAR Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," states:

> 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, cognizant personnel who use and apply engineering analysis and numerical models shall be competent in this 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. Duke Energy will develop and maintain qualification requirements for individuals assigned various tasks. Individuals will be qualified to appropriate job performance requirements per ACAD 98-004. Engineering training guidelines 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.

Regarding qualifications of users of engineering analyses and numerical models:

a. Describe what constitutes the appropriate qualifications for staff and consulting engineers to use and apply the methods and fire modeling tools included in the engineering analyses and numerical models.

b. Describe the process/procedures for ensuring the adequacy of the appropriate qualifications of the engineers/personnel performing the fire analyses and modeling activities.

c. Describe who performed the walk-downs of the MCR and other fire areas in the plant. Describe whether these were the same people who performed the fire modeling analysis.

d. 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 performed adequately and will continue to be performed adequately during post-transition.

e. Explain the communication process between the consulting engineers and MNS personnel to exchange the necessary information and any measures taken to assure the fire modeling was performed adequately and will continue to be performed adequately during post-transition.

Duke Energy Response:

a. Duke Energy considers the following to be appropriate qualifications for Fire Protection Engineers and contractors to perform and review Fire Modeling analyses using Fire Modeling tools and methods:
    - The INPO accredited training program will be used to ensure that individuals are qualified to perform the applicable to the task.
    - The training program will include activities such as completing reading assignments of task instructions (i.e. calculation procedures) for the relevant work that will be performed. This requirement also includes completing independent studies for relevant industry methodology and/or guidance documents such as NUREG/CR-6850, NUREG-1934, NUREG-1805, and other applicable fire modeling user's guide documents, etc.
    - Education on the subject of combustion, fire dynamics and/or fire modeling. Examples of education activities meeting this requirement includes
        - Academic training in fire analysis (e.g., fire modeling, fire dynamics, etc)
        - Demonstration of comprehension and proficiency in fire modeling

   For the specific case of Duke Energy contractors, the contractor's quality assurance process ensures that the personnel performing the fire modeling are qualified and trained. The contractor's qualifications are maintained by the contracting company quality assurance manager, who ensures that the education credentials, appropriate quality assurance training and reading assignments are completed before the tasks are performed.

b. Fire modeling calculations are required to be performed by a Fire Protection Engineer who meets the qualification requirements of Section 2.7.3.4 of NFPA 805. The qualification process is based on the following programs, which provide the minimum training necessary to perform calculations and analyses:
    - Fire Protection Plant Change Impact Review
    - Fire Protection Engineer
    - Basic Fire Modeling

   The requirement in NFPA 805 listed above will continue to be met and adhered to through Duke Energy procedures and project management of contractor support staff. For personnel performing fire modeling or FPRA development and evaluation, Duke Energy maintains qualifications. The qualifications are developed in accordance with Duke's Accredited Training Program. The qualifications 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.

c. Hughes personnel performed the walk-downs and fire modeling analysis for the MCR. For the other fire areas, ERIN personnel performed the initial walk-downs and then applied the Hughes developed GFMTs. Duke personnel conducted subsequent walk-downs.

d. Throughout the NFPA 805 transition process, the Fire Protection Engineers who conducted the fire modeling and the PRA engineers maintained frequent communications and worked together developing the necessary data, documentation, and quantification infrastructure. This process will continue during transition implementation and future established activities as it is based on procedures and a systematic FPRA methodology that is consistently applied throughout the fleet of nuclear plants.

e. Currently, knowledge transfer between the consulting engineers and the station and fleet personnel has been in progress through the development of fire risk insights, the RAI process, updates to the analysis based on plant modifications/changes, and use of the FPRA analysis to support plant activities such as other regulatory activities where the FPRA is required to support.

- The FM/FPRA qualification relevant to fire modeling tasks will be developed and maintained jointly by the Fire Protection and PRA groups.
- Fire modeling personnel will also prepare and maintain the calculations supporting the FPRA analysis.

It should also be noted that portions of the GFMTs calculation and the MCR Abandonment calculation methodology are used at the other Duke Energy nuclear plants.

Probabilistic Risk Assessment (PRA) RAIs - 60-Day Responses

PRA RAI 01.a

Section 2.4.3.3 of NFPA 805 states that the probabilistic safety assessment (PSA) (PSA is also referred to as PRA) approach, methods, and data shall be acceptable to the authority having jurisdiction (AHJ), which is the NRC. Regulatory Guide (RG) 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, Nuclear Energy Institute (NEI) 04-02, Revision 2, as providing methods acceptable to the NRC staff for adopting a fire protection program consistent with NFPA-805. RG 1.200 describes a peer review process utilizing an associated ASME/ANS standard (currently ASME/ANS-RA-Sa-2009) as one acceptable approach for determining the technical adequacy of the PRA once acceptable consensus approaches or models have been established for evaluations that could influence the regulatory decision. The primary result of a peer review are the facts and observations (F&Os) recorded by the peer review and the subsequent resolution of these F&Os.

Clarify the following dispositions to fire F&Os and Supporting Requirement (SR) assessment identified in LAR Attachment V that have the potential to impact the FPRA results and do not appear to be fully resolved:

a) PP-87-01 and PP-C3-01: The resolutions to these F&Os state that "a subsequent walkdown was conducted for plant partitioning and is documented in the Fire Scenario Report." The walkdown sheets provided in Appendix F of the Fire Scenario Report (MCC-1535-0158-003) are principally focused on identifying targets, do not appear to address plant partitioning features, and are all dated prior to the peer review held in September 2009. Provide a description of the subsequent walkdown performed for plant partitioning, the results of this walkdown, and when the walkdown was performed.

Duke Energy Response:

a) The subsequent walkdowns referred to in F&O PP-B7-01 and PP-C3-01 were conducted over several months spanning from October 2009 through June 2010. These were general area walkdowns used to develop and confirm various fire scenarios and to perform the multi-compartment screening analysis. As stated in MCC-1535.00-00-0101, FPRA Partitioning and Frequency Analysis, Section 7.1, the fire compartments (i.e., per NUREG/CR-6850 Section 1.1) were mapped directly to fire areas (i.e., the existing MNS Fire Areas, under the Appendix R program). These subsequent walkdowns verified the integrity of the compartments, as reflected in the Fire Scenario Report (MCC-1535.00-00-0104), Appendix D [MNS Fire Scenario Report Multi-Compartment Screening Analysis]. As reflected in Appendix D, the "Fire Zone Configuration Notes" column, compartment boundaries were confirmed through walkdowns while evaluating multi-compartment fire scenarios.

   As stated in MCC-1535.00-00-0158, the Fire PRA Application Calculation, Table 4-1, in response to F&O PP-B2-01, the fire scenario walkdowns "...evaluate[d] non-rated barriers that are relied on to truncate the zone of influence *and identify targets in exposed compartments from an adjacent exposing compartment*" [emphasis added]. Thus, any partitioning features that failed to protect a target in an adjacent compartment would be reflected within a fire scenario as one or more additional targets, outside of the compartment with the ignition source under evaluation. Since there were no instances found of affecting targets in exposed adjacent compartments, no additional targets were added to the scenarios documented in the Fire Scenario Report, Appendix F [Scenario Walkdown Results].

PRA RAI 04

Section 2.4.3.3 of NFPA-805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA-805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA-805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.174 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes. The NRC staff review of the information in the LAR has identified additional information that is required to fully characterize the risk estimates.

LAR Section 4.5.1.3 has the following statement: "All F&Os that were defined as suggestions have been dispositioned and will be available for NRC review along with the dispositions related to the supplemental F&Os that were generated outside the consensus process." Discuss the process and basis for the development of these "supplemental" F&Os generated outside the consensus process. If they are peer review F&Os, provide the F&O and associated dispositions.

Duke Energy Response:

There is no process for development of new F&Os after closure of the peer review. These items were identified by a specific peer review team member after the peer review exit meeting but before the final peer review report was issued. Although reviewed and discussed by the peer review team, no formal consensus was reached. Based on the lack of consensus, it is Duke Energy's position that the items identified are not peer review F&Os. Duke addressed the items even though they were not peer review F&Os. The associated disposition information is located in MCC-1535.00-00-0158, the MNS FPRA Application Calculation.

PRA RAI 05

Section 2.4.3.3 of NFPA-805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA-805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA-805 based program, and all future plant changes to the program, shall be 'acceptable to the NRC. RG 1.17 4 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes. The NRC staff review of the information in the LAR has identified additional information that is required to fully characterize the risk estimates.

LAR Section 4.5.2.2 provides a high-level description of how the impact of transition to NFPA-805 impacts DID and safety margin was reviewed, including using the criteria from Section 5.3. of NEI 04-02 and from RG 1.205. However, no explanation is provided of how specifically the criteria in these documents were utilized and/or applied in these assessments.

a) Provide further explanation of the method(s) or criteria used to determine when a substantial imbalance between DID echelons existed in the Fire Risk Evaluations (FREs), and identify the type of plant improvements made in response to this assessment.

b) Also, provide further discussion of the approach in applying the NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," Revision 2, (ADAMS Accession No. ML081130188) criteria for assessing safety margin in the FREs.

Duke Energy Response:

a) See "Attachment 1 - PRA RAI 5a Information" (at the end of this Enclosure) starting at section 4.3.2 for DID information. Attachment 1 is an excerpt from the FRE calculation, MCC-1435.00-00-0041, Rev. 2 used in the FRE process for determining the adequacy of DID.

b) Safely Margin Considerations:

Based on NEI 04-02, the requirements related to safety margins for the FRE are described for each of the specific analysis types used in support of the fire risk assessment.

- Fire Modeling
- Plant System Performance
- FPRA Logic Model
- Success Path Verification

1. Fire Modeling

For fire modeling used in support of the FRE (i.e., as part of the Fire PRA), the results were documented as part of the qualitative safety margin review. The following statement regarding Fire Modeling was documented and confirmed for each fire area fire risk evaluation.

Fire modeling performed in support of the transition has been performed within the Fire PRA utilizing codes and standards developed by industry and NRC staff to provide realistic yet conservative results. Specifically, the heat release rates utilized in the transition analysis are based upon NUREG/CR-6850, Appendix E, Severity Factors. These heat release rates are conservative and represent values used to screen out fixed ignition sources that do not pose a threat to the targets within specific fire compartments and to assign severity factors to unscreened fixed ignition sources. The combined analysis approach is used during transition; therefore, maximum expected fire scenario/limiting fire scenario have not been analyzed separately. The bases for the application of these fire modeling codes and standards were not altered in support of this FRE.

2. Plant System Performance

This review documented that the Safety Margin inherent in the analyses for the plant design basis events was preserved in the analysis for the fire event and satisfied the requirements of this section. The following statement regarding Plant System Performance was documented and confirmed for each fire area fire risk evaluation.

Performance parameters were originally established to support nuclear performance criteria contained in the plant specific accident analyses. These analyses established component and system performance criteria necessary to establish safe and stable plant operation in the event of a fire. These performance parameters were not modified as a result of this FRE.

3. FPRA Logic Model

This aspect of Safety Margin was implicitly addressed in the FRE. This treatment was deemed sufficient because the FRE risk quantification method does not impact Safety Margin inherent in FPRA. The FPRA is characterized as having safety margin based on the following characteristics, which satisfy the guidelines established in NEI 04-02 Rev 2:

- Changes to the internal Events PRA in support of the fire PRA logic model development were performed using the same methods and criteria for the internal events PRA model development. In September of 2009, the at-power FPRA was subjected to a Peer Review using the applicable requirements of the Addenda to American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) standard, ASME/ANS RA-S-2008, "Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant", February 2009. The resolution of the resulting Findings and Suggestions is documented in MCC-1535.00-00-0158, Revision 3.

- To ensure that safety margin, inherent in the PRA model is preserved, treatments were applied via methods such as the following:

  - Application of industry recommended fire related event frequencies and probabilities.
  - Increased Human Reliability Analysis failure probabilities with conservative multipliers to account for fire initiating event effects (note that further discussion of this approach will be addressed in the response to PRA RAI1 c).
  - Application of the guidance in NUREG/CR-6850.

4. Success Path Verification (corresponds to Miscellaneous in NEI 04-02)

The "Miscellaneous" category addresses any other analyses not addressed in the three elements discussed above. For MNS, the applicable "miscellaneous" analysis is Success Path Verification, which is the confirmation that changes in CDF and LERF are below the acceptance criteria in order to establish that a success path effectively remains available. NEI 04-02, 5.3.5.3 guidance requires that codes and standards or their alternatives accepted for use by the NRC are met, that safety analyses acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met, or provides sufficient margin to account for analysis and data uncertainty.

PRA RAI 07.a, 07.b, and 07.c

Section 2.4.3.3 of NFPA-805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA-805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA-805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.17 4 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes.· The NRC staff review of the information in the LAR has identified additional information that is required to fully characterize the risk estimates.

LAR Attachment G identifies three categories of operator manual actions [OMA(s)] in post-fire procedures: (1) recovery actions (RAs) to reduce risk, (2) RAs required for DID, and (3) actions associated with VFDRs but are screened out due to no or very low risk and are not considered recovery actions. Provide the following regarding these screened actions:

a) Are the screened OMAs the same as the "pre-existing" OMAs discussed on Page 40 of the LAR? If not, explain the difference.
b) Describe the criteria for screening OMAs as RAs.
c) Discuss how the screened-in OMAs are treated or modeled in the FPRA.

Duke Energy Response:

a) No. The screened OMAs include the "pre-existing OMAs" and also include the "new OMAs" from the new deterministic analysis. Some of the pre-existing OMAs were screened out as PCS actions (retained in Attachment G of the LAR) or required for cold shutdown only. The remainder of the pre-existing OMAs that address cable/component impacts in the fire area were evaluated as VFDRs, and a subset of those screened out based on risk. The deterministic reconstitution analysis identified "new OMAs" to address newly identified cable/component issues and a subset of those screened out based on risk.

b) Please refer to "Attachment 2 - PRA RAI 07 Information" (at the end of this Enclosure) for screening process and criteria.

c) The fire risk evaluation resulted in a population of OMAs that were screened in as recovery actions based on DID or Risk. The action that was screened in based on DID was not modeled as a human failure event in the FPRA. The actions that screened in based on Risk are modeled in the FPRA as human failure events in accordance with applicable human reliability analysis modeling requirements.

PRA RAI 08

Section 2.4.3.3 of NFPA-805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA-805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA-805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.17 4 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes. The NRC staff review of the information in the LAR has identified additional information that is required to fully characterize the risk estimates.

LAR Table S-3, Implementation Item #12 commits to updating the FPRA and re-evaluating the risk results after installation of the plant modifications identified in Table S-2 are completed. This implementation item does not address completion of the implementation items identified in Table S-3. Discuss your plans for re-evaluating the risk results following completion of the Table S-3 implementation items, guidelines for taking action based on the results, and actions that will be taken if the guidelines are exceeded.

Duke Energy Response:

Following installation of modifications, completion of implementation items, and the as-built installation details, additional refinements surrounding the modifications and implementation items will be incorporated into the FPRA model, as required. If the as-built change in risk for a fire area exceeds the estimates reported in LAR Tables W-3 and W-4, the responsible feature will be identified and evaluated per the post transition change process per section 4.7.2 of the LAR.

Review of the implementation items in Table S-3 determined that only item 12 has potential impacts on the fire PRA.

PRA RAI 14

Section 2.4.3.3 of NFPA-805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA-805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA-805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.17 4 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to I these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes. The NRC staff review of the information in the LAR has identified additional information that is required to fully characterize the risk estimates.

LAR Section W.2.1 describes only one method for estimating the delta risk, as follows: "The compliant case was created by manipulating the Fire PRA model to 'remove' the VFDR(s). Fire PRA manipulations involved 'oggling off' or excluding specific PRA basic events to remove the potential fire induced failure associated with the VFDRs." It does not address if there are any exceptions to this method, such as potentially with MCR abandonment scenarios or the use of bounding methods per FAQ 08-0054 (use of the bounding method seems to be implied by the delta risk results presented in Tables W-3 and W-4 for some fire areas). Provide further description of the methods used to determine the change in risk values reported in LAR Tables W-3 and W-4 and additional discussion of the results as requested below.

a) RG 1.174 states that combined change requests (i.e., those that combine risk increases with risk decreases) should report the risk increases and risk decreases separately. Please explain how these values can be obtained from the tables in Appendix W or provide for the post-transition plant an estimate of the risk increase from the retained VFDRs and, separately, the risk decrease associated with modifications made only to reduce risk.

b) Were any methods other than the basic event toggling already described used to determine the fire area change in risk or delta risk reported in LAR Tables W-3 and W-4? If so, describe each method.

c) Describe how the change in risk was determined for MCR abandonment scenarios, including a summary of how the CCDP was determined for the compliant and the variant plants. Note that an overestimate of the compliant plant risk, unless offset with a similar overestimate in the variant plant risk, results in a non-conservative analysis of the delta risk. If the method described applies different assumptions to the variant and the compliant plant risk estimates, an indeterminate but non-conservative impact on the change-in-risk estimate may result.

Duke Energy Response:

a) The risk increases from the retained VFDRs are shown in the "Total" row of LAR tables W-3 and W-4 under the "Fire Risk Eval. ΔCDF" and "Fire Risk Eval ΔLERF" columns. Regarding the portion of the RAI on risk decreases, the Liquid Radwaste System (WL) modification was the only modification credited for risk reduction and only applies to the LERF results. The amount by which the risk would decrease as a result of this modification is found in the "Total" row under the "Offset Risk (WL Mod) Δ LERF" column of tables W-3 and W-4.

b) For most of the VFDR evaluations, basic event toggling was the method used to provide a delta CDF and delta LERF result. In other VFDR evaluations, the use of basic event toggling was investigated, but upon inspection, it was determined that the delta CDF/delta LERF would not be measureable due to factors such as cable routing or conservative deterministic assumptions (e.g., off-site power availability); thus, the VFDR under consideration does not change the plant risk. In certain cases where the function associated with the VFDR was not explicitly modeled, qualitative arguments were provided to support the delta risk conclusion.

c) There are two MCR abandonment scenarios modeled in the FPRA; one for the sub-group of ignition sources that make up the Main Control Board (MCB) fire scenarios and another for the sub-group that comprises the non-MCB fire scenarios. For each case, the Conditional Core Damage Probability (CCDP) was based on the worst case scenario for that sub-group, plus additional failures, to ensure that the Safe Shutdown Facility (SSF) was the sole success path for the abandonment scenario. Similar to other scenarios, to determine the compliant case, these scenarios were re-quantified after removing the non-compliances associated with the VFDRs for the fire area.

PRA RAI 15

Section 2.4.3.3 of NFPA-805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA-805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA-805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.17 4 provides quantitative guidelines on CDF, LERF; and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes. The NRC staff review of the information in the LAR has identified additional information that is required to fully characterize the risk estimates.

LAR Tables W-3 and W-4 report results for total delta LERF of 1.06E-06 per year for Unit 1 and 9.32E-07 per year for Unit 2. These tables also report an "Offset Risk" from crediting a risk reduction modification to the Liquid Waste Recycle System (WL) that results in a substantial reduction in the reported total delta LERF values for both units, yielding a net reduction in delta LERF from the transition to NFPA-805. Given the importance of this modification to the transition, provide a description of how the risk reduction from this modification was calculated. The response should include a discussion of key assumptions and non-conservatisms, and the impact of any non-conservatisms on the reported "Offset Risk."

Duke Energy Response:

The WL system collects condensate from various ventilation system drains in containment. The condensate drains to the condensate drain tank in the Auxiliary Building via a 6 inch line that is normally open. This pathway presents a LERF concern if the pathway is not isolated due to random or accident induced failures and containment pressure is high enough to blow out the loop seal. The risk associated with this line is removed by restricting the flow area with an engineering solution, such as an orifice, to a point where it no longer presents a LERF concern. The risk offset was determined by solving the model with and without the WL line isolation failure in the LERF estimation. The risk reduction quantification assumes 100% reliability of the modification as the modification is the installation of a passive device whose failure to reduce flow is regarded as negligibly small. Some conservatism that is included in the offsite consequence analysis is that no credit is assumed for reduction in the fission product release by either the ventilation unit condensate drain tank or the auxiliary building structure.

PRA RAI 18

Section 2.4.3.3 of NFPA-805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the staff for adopting a fire protection program consistent with NFPA-805. Methods that have not been determined to be acceptable by the NRC staff or acceptable methods that appear to have been applied differently than described require additional justification to allow the NRC staff to complete its review of the proposed method.

The MCB is described as having a horseshoe arrangement that is fully enclosed and is effectively a sub-enclosure. The analysis of MCB fires treats the front and back panels of the horseshoe as an integral part of the MCB.

a) FAQ 14-0008 provides guidance on how MCB fires should be treated for MCBs that are sub-enclosures. Describe how your MCB configuration and MCB fire scenario analysis is consistent with the FAQ.

b) Describe how MCB fire scenarios are postulated and evaluated, including how the fire ignition frequency is determined for each scenario, how NUREG/CR-6850 Appendix L is applied to individual scenarios, how partitions between panels/cabinets are treated if credited, and how propagation between the front and back sides of the MCB is evaluated including identification of and evaluation of damage to target sets.

Duke Energy Response:

a) FPRA FAQ 14-0008 was developed after submittal of the MNS NFPA 805 LAR. The MCB fire scenario development for MNS is consistent with many aspects of the FAQ; however, it does not fully address the front-to-back considerations for application of the methodology in Appendix L of NUREG/CR-6850. The MNS FPRA will be updated to address the front-to-back considerations. Any deltas in the risk results will be captured in the response to PRA RAI 3. The following describes how the remaining aspects of the MNS MCB analysis is consistent with the FAQ.

    The rear side of the MCB is classified as an integral part of the MCB because the rear and front sides are connected together as a single enclosure. There is a continuous

overhead ceiling connecting the front and back sides. Therefore, the methodology in FPRA FAQ 14-0008 is considered applicable. The MNS MCB fire scenario analysis is consistent with the FAQ's requirements guidance for crediting partitions. For basis, see discussion in the "MCB Fire Ignition Frequency" discussion in Part b of this RAI response.

b) The key elements of the MCB fire scenario development are described below:

MCB Fire Ignition Frequency:

The entire frequency for the MCB for a single unit was applied to each of the MCB Fire scenarios. Application of the entire ignition frequency captures the potential addition of risk from the rear of the MCB panel in accordance with Alternative 2 of FAQ 14-0008. There are at least 4 separate, standalone MCB cabinets, however, no partitions were credited to reduce frequency.

MCB Severity Factor and Non-Suppression Factors:

Severity factor/Non-Suppression probability was based on NUREG/CR 6850 Appendix L , Figure L-1, " Likelihood of Target Damage Calculated as the Severity Factor Times the Probability of Non-suppression for MCB Fires." Values selected were based on the distribution for qualified cables. Appendix L probabilities were credited based on distances between the instrumentation and controls for major functions on the control board.

MCB Fire CCDP:

The approach involved calculating the cumulative CCDP as the fire spreads from one end of an MCB segment to the other. In some cases, the entire set of targets for an MCB cabinet was assumed for a given scenario. In other cases, the distance between key failures dictated that multiple scenarios for a given MCB cabinet be postulated.

PRA RAI 21

Section 2.4.3.3 of NFPA-805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the NRC staff for adopting a fire protection program consistent with NFPA-805. Methods that have not been determined to be acceptable.by the NRC staff or acceptable methods that appear to have been applied differently than described require additional justification to allow the NRC staff to complete its review of the proposed method.

The licensee's analysis dispositions how each MSO from the industry generic list was addressed in the FPRA model. This list is not consistent with the generic MSO list in Appendix G of NEI 00-01, Rev. 2 which, according to Attachment F of the LAR, may not have been the source of the generic MSO list used in the FPRA. Identify the MSOs in Appendix G of NEI 00-01, Rev. 2, that are not identified in Table A-1 of the Fire Model Development Report and describe how these are dispositioned in the FPRA model. Provide justification for any generic MSOs not identified in Table A-1 or provide updated risk results as part of the aggregate change-in-risk analysis requested in PRA RAI 03 that incorporates the additional MSOs in the FPRA model.

Duke Energy Response:

Note: See Related SSA RAI 3.

The MSOs listed in the MNS FPRA are based on those listed in Attachment A of the MNS MSO expert panel calculation, MCC-1435.00-00-0023. In response to a similar RAI, SSA RAI 3, MNS MSO expert panel calculation was updated by MNS Engineering to provide a comparison and reconciliation of the original MSO list in MCC-1435.00-00-0023 with the MSO lists in Revision 2 (and 3) of NEI 00-01. Review of NEI 00-01, Revision 3, MSO list was completed, and a response to each new item is included in the Attachment B to the revised MSO calculation (MCC-1435.00-00-0023, Rev. 2). No additional MSOs were required to be added to the FPRA as a result of this review.

Table 1 - RG 1.174 Acceptance Criteria

Region	ΔCDF/rx-yr	ΔLERF/rx-yr	Status	Comments/Conditions
I	≥ 1.0E-05	≥ 1.0E-06	Unacceptable	Proposed changes in this region are not acceptable, regardless of baseline CDF and LERF.
II	< 1.0E-05 and ≥ 1.0E-06	< 1.0E-06 and ≥ 1.0E-07	Acceptable w/ conditions	Proposed changes in this region are acceptable provided the cumulative total CDF from all CDF initiators is less than 1.0E-04/rx-yr and from all LERF initiators is <1E-05/rx-yr. Cumulative effect of changes must be tracked and included in subsequent changes.
III	< 1.0E-06	< 1.0E-07	Acceptable w/ conditions	Proposed changes in this region are acceptable provided the cumulative total CDF from all initiators is less than 1.0E-03/rx-yr and from all LERF initiators is <1E-04/rx-yr. Cumulative effect of changes must be tracked and included in subsequent changes.

In order to ensure the criteria above were met cumulatively for the plant, the acceptance criteria for an individual Fire Area was initially established as delta CDF less than 1E-07/rx-yr and delta LERF less than 1E-08/rx-yr. These acceptance criteria are intended to support a total plant delta CDF and delta LERF within the acceptance guidelines of RG 1.174, delta CDF less than 1E-06/rx-yr and delta LERF less than 1E-07/rx-yr, for plant total CDF/LERF (conservatively including internal events contribution to plant risk) of 1E-4/1E-5/rx-yr, respectively.

Also, to ensure that the acceptance criteria were not solely based on low ignition frequency, the CCDP values for each of the post-transition baseline case scenarios were reviewed.

4.3.2 Defense-in-Depth (DID)

4.3.2.1 Guidance

A review of the impact of the change on DID was performed, using the guidance below from NEI 04-02. NFPA 805 defines defense-in-depth as:

- Preventing fires from starting
- Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting damage
- Providing adequate level of fire protection for structures, systems and components important to safety; so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

In general, the DID requirement was satisfied if the proposed change does not result in a substantial imbalance among these elements (or echelons). The review of DID was qualitative and addressed each of the elements with respect to the proposed change. Fire protection features

and systems relied upon to ensure DID were identified in the assessment (e.g., detection, suppression system).

Consistency with the DID philosophy is maintained if the following acceptance guidelines, or their equivalent, are met:

- A reasonable balance is preserved among 10 CFR 50.48(c) DID elements.
- Over-reliance and increased length of time or risk in performing programmatic activities to compensate for weaknesses in plant design is avoided.
- Pre-fire nuclear safety system redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences of challenges to the system and uncertainties (e.g., no risk outliers). (This should not be construed to mean that more than one safe shutdown/NSCA train must be maintained free of fire damage.)
- Independence of DID elements is not degraded.
- Defenses against human errors are preserved.
- The intent of the General Design Criteria in Appendix A to 10 CFR Part 50 is maintained.

4.3.2.2 DID Process

Each Fire Area was evaluated for the adequacy of DID. In accordance with NFPA 805 Section 2.4.4, Plant Change Evaluation, "The evaluation process shall consist of an integrated assessment of the acceptability of risk, DID, and safety margins." NFPA 805 Section 4.2.4.2 refers to the acceptance criteria in this section. Therefore fire protection systems and features required to demonstrate an adequate balance of DID are required by NFPA 805 Chapter 4.

Considerations for determining what constitutes DID criteria are broken down into three echelons, which are provided in Table 2. These echelons cover a wide range of administrative, active and passive systems and/or features that are qualitatively reviewed against the particular risk characteristics of a fire area where their incorporation would further provide necessary mitigation effects.

Balance of these three DID echelons is inherent to the process. There are fundamental fire protection features in every fire protection program that will be required to meet NFPA 805 Chapter 3. These features include hot work and combustible controls, pre-fire plans, rated barriers, etc. The FRE utilizes this approach to ensure these fundamental features are always required to meet NFPA 805 Chapter 4. Figure 7 shows the relationship between DID echelons and fundamental requirements of NFPA 805 Chapter 3.

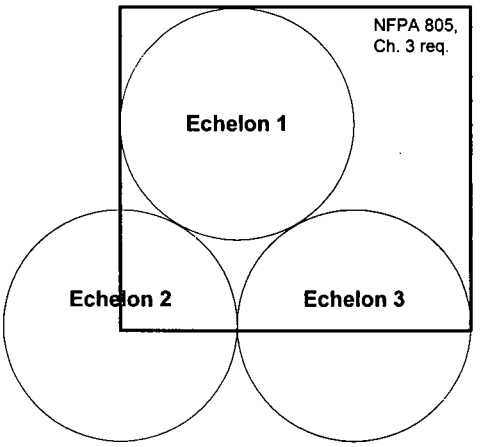


Figure 7 – Fundamental DID Balance

Table 2 - Considerations for Defense-in-Depth Determination

Method of Providing DID	Considerations
Echelon 1: Prevent fires from starting	
▪ Combustible Control ▪ Hot Work Control	Combustible and hot work controls are fundamental elements of DID and as such are always in place. The issue to be considered during the FREs is whether this element needs to be strengthened to offset a weakness in another echelon thereby providing a reasonable balance. Considerations include: ▪ Creating a new Transient Free Areas ▪ Modifying an existing Transient Free Area The fire scenarios involved in the FRE quantitative calculation should be reviewed to determine if additional controls should be added. Review the remaining elements of DID to ensure an over-reliance is not placed on programmatic activities to compensate for weaknesses on plant design.
Echelon 2: Rapidly detect, control and extinguish promptly those fires that do occur thereby limiting fire damage	

Table 2 - Considerations for Defense-in-Depth Determination

Method of Providing DID	Considerations
▪ Detection system ▪ Automatic fire suppression ▪ Portable fire extinguishers provided for the area ▪ Hose stations and hydrants provided for the area ▪ Fire Pre-Fire Plan	Automatic suppression and detection may or may not exist in the Fire Area of concern. The issue to be considered during the FRE is whether installed suppression and or detection is required for DID or whether suppression/detection needs to be strengthened to offset a weakness in another echelon thereby providing a reasonable balance. Considerations include: ▪ If a Fire Area contains both suppression and detection and fire fighting activities would be challenging, both detection and suppression may be required ▪ If a Fire Area contains both suppression and detection and fire fighting activities would not be challenging, require detection and manual fire fighting (consider enhancing the pre-plans) ▪ If a Fire Area contains detection and a recovery action is required, the detection system may be required. ▪ If a Fire Area contains neither suppression nor detection and a recovery action is required, consider adding detection or suppression. The fire scenarios involved in the FRE quantitative calculation should be reviewed to determine the types of fires and reliance on suppression should be evaluated in the area to best determine options for this element of DID.
Echelon 3: Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed	
▪ Walls, floors ceilings and structural elements are rated or have been evaluated as adequate for the hazard. ▪ Penetrations in the Fire Area barrier are rated or have been evaluated as adequate for the hazard. ▪ Supplemental barriers (e.g., ERFBS, cable tray covers, combustible liquid dikes/drains, etc.) ▪ Fire rated cable ▪ Reactor coolant pump oil collection system (as applicable) ▪ Guidance provided to operations personnel detailing the required success path(s) including recovery actions to achieve nuclear safety performance criteria.	If fires occur and they are not rapidly detected and promptly extinguished, the third echelon of DID would be relied upon. The issue to be considered during the FRE is whether existing separation is adequate or whether additional measures (e.g., supplemental barriers, fire rated cable, or recovery actions) are required offset a weakness in another echelon thereby providing a reasonable balance. Considerations include: ▪ If the VFDR is never affected in the same fire scenario, internal Fire Area separation may be adequate and no additional reliance on recovery actions is necessary. ▪ If the VFDR is affected in the same fire scenario, internal Fire Area separation may not be adequate and reliance on a recovery action may be necessary. ▪ If the consequence associated with the VFDRs is high regardless of whether it is in the same scenario, a recovery action and / or reliance on supplemental barriers should be considered. ▪ There are known modeling differences between a FPRA and nuclear safety capability assessment due to different success criteria, end states, etc. Although a VFDR may be associated with a function that is not considered a significant contribution to CDF, the VFDR may be considered important enough to the

Table 2 - Considerations for Defense-in-Depth Determination

Method of Providing DID	Considerations
	NSCA to retain as a recovery action. The fire scenarios involved in the FRE quantitative calculation should be reviewed to determine the fires evaluated and the consequence in the area to best determine options for this element of DID.

4.3.2.3 Defense-in-Depth - Recovery Action Considerations

Reliance on Recovery Actions in lieu of protection is considered part of the third echelon of DID. Per NFPA 805, recovery actions are defined as: "Activities to achieve the nuclear safety performance criteria that take place outside of the main control room or outside of the primary control(s) station for the equipment being operated, including the replacement or modification of components."

If the VFDR is characterized as a 'Separation Issue', and the change in risk (delta CDF and delta LERF) is acceptable, a recovery action can be considered as a means to provide an adequate level of DID. The 'additional risk presented by the use of the recovery action' would be characterized as the calculated change in risk of the 'Separation Issue'.

4.3.3 Safety Margin Assessment

A review of the impact of the change on safety margin was performed. The guidelines for making that assessment are summarized below.

- Codes and standards or their alternatives accepted for use by the NRC are met, and
- Safety analysis acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met, or provides sufficient margin to account for analysis and data uncertainty.

The requirements related to safety margins for the change analysis is described for each of the specific analysis types used in support of the fire risk evaluation.

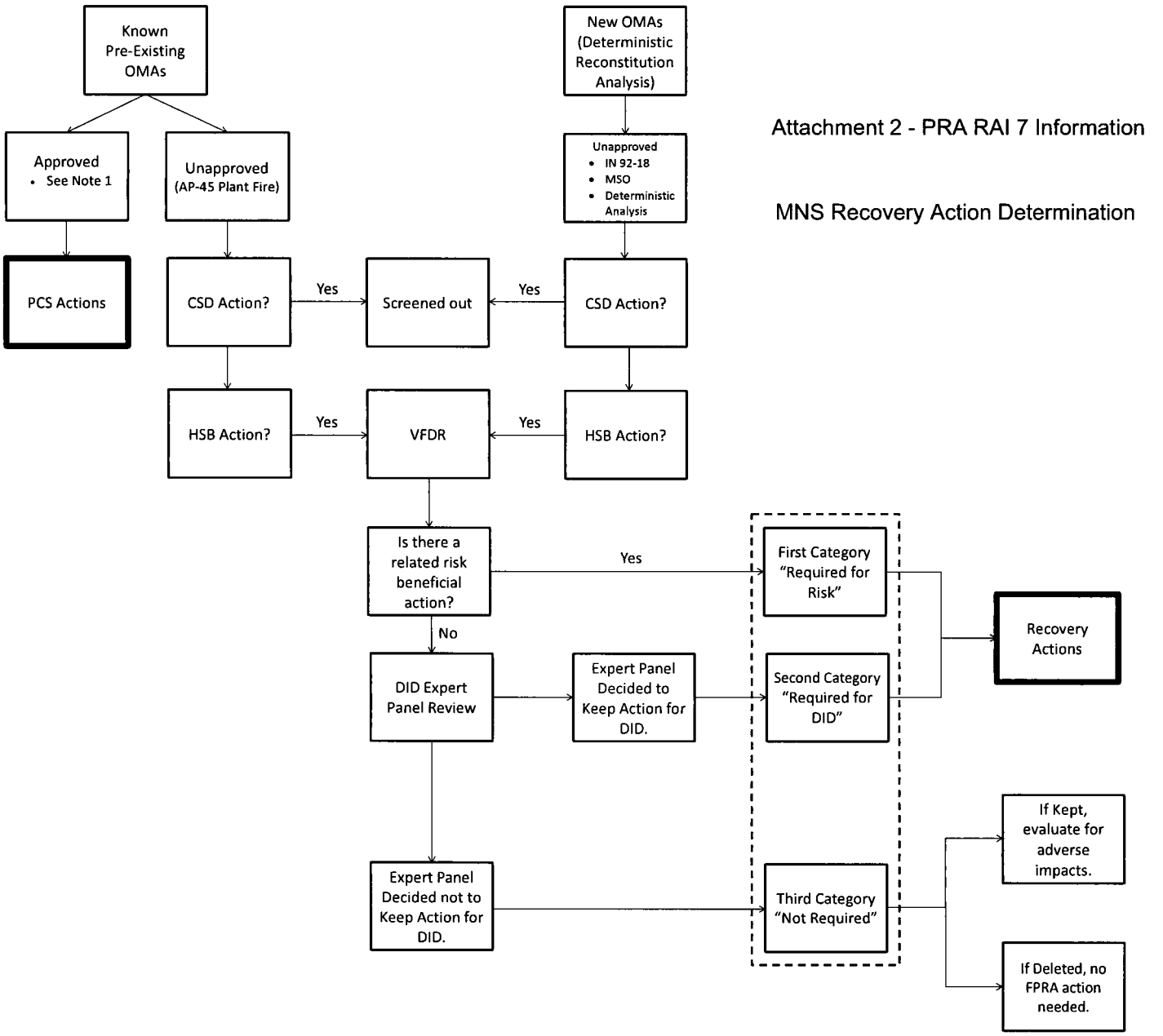
These analyses can be grouped into four categories. These categories are:

- Fire Modeling
- Plant System Performance
- FPRA Logic Model
- Success Path Verification

The following guidance on these topics is provided. Additional information is contained in NEI 04-02 Section 5.3.5.3.

Attachment 2 - PRA RAI 7 Information

MNS Recovery Action Determination



Note 1 | For example SSF activation via AP-24 (Safety Evaluation Report, NUREG-0422, "Transient Analysis of the Research Reactor MARIA MC Fuel Elements Using RELAP5 Mod 3.3," Supplement 6.)