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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, ADAMS Accession Number ML14016A097.
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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6. MNS Letter, Response to August 28, 2014 NRC Request for Additional Information Regarding License Amendment Request To Implement A Risk-Informed Performance-Based Fire Protection Program, dated October 13, 2014, No ADAMS Number.

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 provided the 60-day RAI responses by letter dated October 13, 2014 (Reference 6). Duke Energy's responses to the 90-day RAIs are provided in Enclosure 1. As a result of some of the responses to the 90-day RAIs, it will be necessary to perform some heat release rate reanalysis and to revise some pages of the LAR. Those LAR revisions and analyses, which are described in the applicable responses, have been entered into the MNS Corrective Action Program. The LAR revisions will be included in the submittal providing the responses to the 120-day RAIs. Responses for the 120-day RAIs will be provided by December 12, 2014.

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 November 12, 2014.

Sincerely,

Steven D. Capps

Enclosure 1

xc:

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ENCLOSURE 1

Duke Energy Responses To The August 28, 2014 90-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:

Fire Protection Engineering (FPE) Request for Additional Information (RAI) - 90-Day Responses

FPE RAI 03

LAR Attachment I, Table 1-1 "Definition of Power Block" states that structures required to meet the radioactive release criteria described in Section 1.5 of NFPA-805 but not required to meet the nuclear safety criteria are not defined within the power block. Currently, the endorsed guidance of Nuclear Energy Institute (NEI) 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Program Under 10 CFR 50.48(c)," states that, where used in Chapter 3, "power block" and "plant" refers to structures that have equipment required for nuclear plant operations, such as containment, auxiliary building, service building, control building, fuel building, radiological waste, water treatment, turbine building, and intake structure, or structures that are identified in the facility's current license basis. As currently described in the LAR Attachment E, the Rad Waste Facility is a standalone building within the Yard Fire Area. Additionally, the Contaminated Material Handling and Waste Handling areas are described as part of the Auxiliary Building. Included in this compartment are Building 1202 and the Waste Solidification Building.

Provide clarification that those structures listed within the guidance are accounted for as either within or not within the power block.

Duke Energy Response:

The McGuire Nuclear Station (MNS) definition of power block was developed based on the guidance provided in FAQ 06-0019 Revision 4 (ML073060545) and the NRC closure memo to FAQ 06-0019 (ML080510224). The MNS definition of power block was developed with respect to those structures required to meet the NFPA 805 nuclear safety performance criteria. This includes structures with the potential to affect power plant operations, the potential to affect equipment important to nuclear safety, and the potential to affect the ability to safely shutdown the plant in the event of a fire.

The NRC closure memo includes a chronological history of the development of FAQ 06-0019 that supports this definition. The NRC Staff comments on FAQ 06-0019, Revision 3 states:

> *"The letter and the intent of the NFPA 805 definition for "power block" and "plant" ("Structures that have equipment required for nuclear plant operations") includes all equipment needed to generate electricity (main turbine, feedwater, circulating water, service water, main steam, etc.) as well as that equipment needed to mitigate accidents required by the Technical Specifications (safety injection, emergency diesel generators, containment spray, emergency service water, etc.)."*

The MNS definition of Power Block as found in LAR Attachment I, Table I-1 is consistent with the NRC Staff Comments on the definition of power block.

Referring to the structures listed in the guidance (FAQ 06-0019), the MNS power block includes Containment (Reactor Buildings), Auxiliary Building (where Nuclear Safety Capabilities Assessment [NSCA] equipment is located), Service Building, Fuel Building, Turbine Building, intake (and discharge) structures, in addition to structures specific to MNS – the Doghouses,

Standby Shutdown Facility (SSF), and yard areas where equipment required to meet the nuclear safety performance criteria goals is located.

MNS does not have a defined Control Building. The Control Room and supporting rooms that may be typically found in a Control Building are located in the Auxiliary Building. The Water Treatment Facility and Switchyard are not included in the power block definition. The Water Treatment Facility is not required to meet the nuclear safety performance criteria, and the Switchyard is not included as the NFPA 805 analysis boundary begins at the Main and Auxiliary Transformers.

Radiological waste areas are not included in the MNS power block definition. There is a series of structures identified in LAR Attachment E, Radioactive Release Transition, which are not identified in the MNS power block. The MNS radiological release areas are not included based on the guidance in the NRC closure memo in that the areas are not needed to generate electricity or needed to mitigate accidents as required by the Technical Specifications. These specific MNS areas include:

- The Rad Waste Facility (located in the Yard)
- The Contaminated Material and Waste Handling areas (Building 1202 and the Waste Solidification Building) which are considered part of the Auxiliary Building
- The Equipment Staging Building (located in the Yard adjacent to the Unit 2 Reactor Building)
- The Compacted & Central Waste Facilities (located in the Yard)
- The Radiography Facility (located in the Yard)
- Warehouse 7 (located outside of the protected area and within the owner controlled area)

FPE RAI 09

In LAR Attachment A, Table B-1, Section 3.3.5.3., the LAR indicates that electrical cables comply with IEEE-383 flame propagation testing (Institute of Electrical and Electronics Engineers Standard 383 "IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations"). The staff noted that the LAR only describes armored cables in conjunction with a discussion of an outer jacket, but the licensee's analysis includes unjacketed armored cables and the staff notes that rapid and significant flame spread is associated with unjacketed armored cables.

a. Describe whether unjacketed armored cable is installed, and if it is, describe the extent and installed locations.

b. Describe the qualification of unjacketed cables and, if this configuration is unqualified, describe how the lack of qualification has been addressed, including in the performance-based analyses.

c. If the unjacketed cables are unqualified, describe the impact on the Fire Probabilistic Risk Assessment (PRA) analysis.

Duke Energy Response:

a. Unjacketed armored cable is installed at MNS, primarily in Containment. A review of the cable database shows that approximately 90 percent of unjacketed armored cable is found in the Unit 1 and Unit 2 Reactor Buildings. The remainder of the armored unjacketed cables is distributed throughout MNS in the following locations: Auxiliary Building (Cable Rooms, Battery Room, ETA and ETB Switchgear and Penetration rooms, etc.), Unit 1 and 2 Turbine Buildings, and Service Building.

b. Procurement of armored cable for use in Duke Energy nuclear power generating stations has always been performed using criteria from cable specifications to ensure a certain level of cable performance and quality.

The term "qualified" is interpreted to mean "cable that meets or exceeds the performance requirements specified in IEEE 383-1974." Because a portion of the cables installed at MNS pre-dates IEEE 383, not all cable was procured to this standard. Power, controls and instrumentation cables purchased today (and since 1981) are required to meet IEEE 383. As stated in the Fire Protection Design Basis Document (MCS-1465.00-00-0008):

> Cable used at McGuire, classified as either power, control or instrumentation, passes the IEEE No. 383-1975 [1974] Flame Test.

The staff has noted "rapid and significant flame spread is associated with unjacketed armored cables," reflecting unanticipated observations made during a series of cable tests performed in 2006, at the Intertek Testing Services Inc. facility in Elmendorf, Texas (formerly Omega Point Laboratories, Inc.). It is important to note that this series of tests was conducted to evaluate fire-induced circuit failure, not flame propagation. Therefore, these tests were performed under conditions that were significantly more severe than testing required to meet IEEE 383 [IEEE Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations] and IEEE 1202 [IEEE Standard for Flame-Propagation Testing of Wire and Cable].

Calculation DPC-1435.00-00-0009, "Performance Characteristics of Duke Armored Cables Under Fire Exposure", is the AREVA Engineering Information Record document EIR 51-9160514-000, an analysis of the same name. As stated in this calculation:

> In the fall of 2007, Duke requested that General Cable Corporation (GCC), the manufacturer of the Duke-specific armored cable that was used in the Duke Armored Cable Control Circuit Tests conducted in 2006, provide IEEE- 1202 test results for both jacketed and unjacketed versions of the same armored cable type used in some of the Duke Armored Cable Control Circuit Tests.
>
> ...These standardized test results indicate that both jacketed and unjacketed version of the armored 8-conductor #12AWG cable passed the IEEE-1202 cable flame propagation test... [bounding the requirements of the IEEE-383 flame test.]
>
> ...The test results also indicate that there is essentially no difference in the flame propagation performance of jacketed and unjacketed applications of this armored cable type when tested in accordance with the IEEE 1202 standard.
>
> Additionally, the IEEE 1202 test report for the jacketed version of cable type tested on 09/17/2007 (Duke GSI Armored 8-conductor #12 1KV-FR- XLPE) noted that flame did in fact "shoot" out the bottom of the sample during the second half of the 20-minute flame exposure test. This indicates a similar phenomenon to what occurred in a few of the Duke Armored Cable Control Circuit Tests conducted at Intertek Laboratories in 2006...

This calculation concludes:

> ...the "shooting flame" condition is not a result of flame propagation internal to the armored cable. Fire testing shows that when flames have occurred at the ends of armored cables, the mechanism that causes flame to occur at this location is

attributed to a series of events and conditions that lead to hot gases and vapors traveling inside the armored shielding from the vicinity of the fire exposure to the open ends of the cable where these gases ignite when mixed with available air...

...Therefore, no additional guidance or conservatism needs to be added to the FPRA based on the use of armored cable and the potential for the "shooting flame" condition to occur.

...In summary, the "shooting flame" phenomenon is not new to armored cables. Fire testing as far back as 1978 suggests that this condition has occurred under certain test parameters. Evidence of this phenomenon has also occurred at times during standardized tests for determining the flame propagation characteristics of armored cables. In no instance has a testing authority ever deemed the results of a standardized flame propagation test for armored cables unacceptable due to this phenomenon.

... Many of the armored cable types use[*d*] at Duke nuclear power generating stations have undergone standardized testing to determine their flame spread propagation characteristics. These tests were typically performed in accordance with the IEEE 383 and/or IEEE 1202 test standards. The results of this testing confirmed that the armored cable types tested are IEEE 383 and/or IEEE 1202 "qualified".

...There are standardized test methods for measuring and determining a cable's flame propagation qualities. IEEE 383 and IEEE 1202 are two such standards that have been deemed acceptable to the NRC. The armored cable types used at Duke nuclear power generating stations have either been qualified to one of these standards or are considered equivalent by comparison.

c. The unjacketed armored cables used at MNS have either been purchased to meet these standards, have been shown to meet the requirements of IEEE 383 or IEEE 1202 through testing, or are considered equivalent to IEEE 383 or IEEE 1202 qualified cables by comparison. Therefore, there is no impact on the Fire PRA analysis.

A revision to LAR Attachment A, Section 3.3.5.3 is planned to be submitted with the 120 day RAI responses to clarify MNS cable is IEEE 383 or equivalent in accordance with flame propagation tests as outlined in FAQ 06-0022.

Safe Shutdown Analysis (SSA) RAIs - 90-Day Responses

SSA RAI 01

LAR Attachment B, Table B-2, identifies certain attributes of NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis" Revision 1, as "Aligns with Intent." For the following attributes, the alignment basis does not fully explain why there are deviations from the recommendations of the attribute.

For each attribute listed below, provide a detailed justification as to what specifically does not align.

a. 3.1C Spurious Operation

b. 3.1.1.7 Offsite power

c. 3.1.1.11 Multiple units

d. 3.2.1.6 Spurious components

e. 3.3.1.3 Isolation Devices

f. 3.3.1.6 Auto Initiation Logic

g. 3.3.1.7 Circuit Coordination

h. 3.5.1.3 Duration of Circuit Failures

i. 3.5.2.1 Circuit Failures Due to an Open Circuit

j. 3.4.1.4 Manual Actions

Duke Energy Response:

a. MNS aligns with the guidance of considering spurious operations, but aligns with the intent for high/low pressure interfaces. 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 Loss of Coolant Accident (LOCA). RG 1.189 Revision 2 Section 5.3.2.c endorses NEI 00-01 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.

b. MNS does not credit offsite power in the deterministic analysis and does not demonstrate it to be free of fire damage. However, MNS analyzed safe shutdown success paths as shown on the functional logic diagrams with and without offsite power for normal and alternate shutdown components. For example, for normal shutdown, pressurizer heaters powered from offsite power need to be off, but spurious operation could turn them on with offsite power available and operator action to trip the supply breakers is required. Without offsite power this is not an issue. Similarly, for alternative shutdown areas, MNS specifically assumes it is available where availability could adversely affect alternative safe shutdown, otherwise, alternative shutdown is shown without offsite power. Thus MNS aligns with the intent of considering offsite power during the analysis but does not credit it.

c. MNS has some common Fire Areas (FA) (e.g., FA-4, 13, and 14), and both units were analyzed for effects of fires in these areas. In unit specific FAs (e.g., FA-2, 5, 19, and 20), there is potential for an opposite unit impact (e.g. in FA-2- Unit 1 Motor Driven Auxiliary Feedwater Pump Room, FA-5- U1 A Train Emergency Diesel Generator, FA-19- Unit 1

Cable Room, and FA-20- Unit 2 Cable Room) that could possibly affect the opposite unit. Unit specific fire areas were mostly found to not have any potential for an opposite unit impact through analysis of fire impacts by equipment and cable location. However, an example of an opposite unit affect concerns the Unit 1, A Train battery charger, which is normally aligned to Unit 1. If a fire occurred in a fire area during those cases that affected the battery charger power supply, operators can promptly realign the battery charger to Unit 2. Since multiple unit effects of fires were analyzed, MNS aligns with the guidance.

d. MNS aligns since equipment was identified for NSCA that could spuriously operate or mal-operate, which meets the NEI 00-01 guidance.

e. Isolation devices were analyzed for the credited train in a given FA; therefore, MNS aligns with the guidance.

f. In addition to the guidance, the analysis was not limited to "...the fire-induced failure of automatic logic circuits.." The automatic interlock signals were analyzed and assumed to occur / not occur in the worst case situation unless specifically analyzed not to do so. This approach exceeds the guidance. Thus, MNS aligns with the guidance.

g. Fault coordination impacts were analyzed for the credited trains/busses. Thus, MNS aligns with the guidance in that a breaker coordination analysis was performed.

h. MNS did not take credit to clear (i.e. the duration of the hot short was not limited) spurious operations in the deterministic analysis. Thus, MNS aligns with this guidance.

i. MNS analyzed open circuits per the NEI 00-01 guidance, including potential high voltage current transformer (CT) secondary damage as itemized. Thus, MNS aligns with the guidance.

j. Variations From Deterministic Requirements (VFDRs) resolutions included in the performance based Fire Risk Evaluations (FREs) included recovery actions as potential mitigating actions to maintain a safe and stable condition for the operational effects of fire damage. Recovery actions are demonstrated to be feasible in accordance with NRC requirements as documented in calculation MCC-1435.00-00-0045. Thus, MNS Aligns with the guidance.

SSA RAI 03.a

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:

a. In describing the documents used for guidance by the expert panel, LAR Attachment F stated that some of these documents (NEI 00-01, NEI 04-06, Fire PRA Task Instruction, and pressurized-water reactor owners group (PWROG) MSO list) were identified as "draft." Describe what reconciliation was done to ensure completeness of the analysis with the final documents.

Duke Energy Response:

a. The documents referenced were used as guidance at the time of convening the expert panel.

The MSO evaluation in MCC-1435.00-00-0023, Rev. 1, "NFPA 805 Transition Expert Panel Report for Addressing Potential McGuire Multiple Spurious Operations (MSO)",

was done against the PWROG draft MSO list. A reconciliation review to NEI-00-01, Revision 3, was completed as part of responding to this RAI. The review looked at the changes from the PWROG MSO list to the MSO list in NEI 00-01, Revision 3, and concluded that all changes are covered in existing MSOs in the MSO calculation. The MSO calculation has been revised to add an attachment that provides a comparison between the new MSO list in NEI 00-01, Revision 3, and the PWROG MSO list that was in the MSO calculation when the LAR was written. The attachment includes a summary of the reconciliation of differences.

The MSO process for the expert panel used the guidance in the Fire PRA Task Instruction (Draft C) and NEI 04-06 (Draft L). The expert panel was completed using these documents as guidance. NEI 04-06 has not been officially issued, and the Draft L has not been changed. NEI 04-06 Draft L was a guide for conducting a self-assessment and that process was considered similar to doing the MSO Expert Panel. Since NEI 00-01 now incorporates the process for the MSO expert panel, the NEI 04-06 process is moot. The MSO process that was in the Fire PRA Task Instruction is also now incorporated into NEI 00-01, Revision 3. A review indicates that the procedure for the MSO expert panel did not change sufficiently to require a change to the expert panel results.

SSA RAI 04

LAR Attachment G, "Recovery Actions Transition," identifies a "third" category of recovery actions (beyond risk reduction and defense-in-depth) as additional actions that screened out due to no or very low risk. The LAR stated that these actions are not considered recovery actions for NFPA-805 and therefore, feasibility is not evaluated against the criteria in NFPA-805 Section B.5.2(e), NEI 04-02, and FAQ 07-0030, "Establishing Recovery Actions."

a. Provide a detailed description of these recovery actions, including:
   - how they were originally identified;
   - what nuclear safety performance goals they are associated with;
   - what fire safe shutdown function they provide; and
   - whether they are currently listed in LAR Attachment G.

b. Describe whether these recovery actions will remain in the procedures. If they will remain in the procedures, justify why feasibility evaluations are not performed for these actions. 1

c. Provide examples of these types of recovery actions.

Duke Energy Response:

NOTE: These SSA RAI 04 responses are complementary to the PRA RAI 07 responses.

NOTE: The information discussed in Attachment G of the LAR is a summary level description of the three categories of VFDR resolutions. The process and individual FA evaluations are in MCC-1435.00-00-0041 "NFPA 805 Transition Risk-Informed, Performance-Based Fire Risk Evaluations." This FRE calculation has individual FA attachments that include the detailed / individualized information, which was not carried forward into Attachment G of the LAR.

a. This "third" category is not a category of recovery actions but a third category of "actions." LAR Attachment G, Step 2, indicates that the population of recovery actions are those that

are required to resolve VFDRs. The population of VFDRs is from the deterministic analysis for each fire area. No VFDRs were written against pre-existing operator manual actions (OMAs) by themselves; there had to be a fire effect on the component. Some component fire impacts that resulted in a VFDR were found to have pre-existing OMAs that could be used to address the VFDR resolution. Each VFDR (See LAR Attachment C) was then evaluated using the performance-based approach, and required recovery actions were identified where necessary to satisfy risk or defense-in-depth (DID). The FRE process, using the Fire PRA, determined that some of these had no or very low contribution to risk and were not required to be carried forward as recovery actions for risk or DID. If VFDRs in Attachment C contained recommended operator actions for the deterministic analysis, but did not result in required recovery actions, then the recommended action was not required to be implemented and not reflected in Attachment G.

Some recommended actions are designated as Primary Control Station (PCS) actions, which are not Recovery Actions. PCS actions are required to be performed but do not require the feasibility assessment that Recovery Actions do. Attachment G lists all the PCS actions and Recovery Actions (either for Risk or DID).

LAR Attachment C has a discussion for each VFDR including performance goal and function. The VFDRs of this "third" category are identified with a disposition of "Satisfies Risk, DID, and Safety Margin Criteria Without Further Action."

b. These non-required actions are strong candidates for deletion from fire response procedures. They may be desired to be retained by OPS as potential margin items or otherwise. If maintained, see PRA RAI 07d.

c. Examples of these "third" category actions where VFDRs did not result in required recovery actions but are presently in fire protection procedures are:

- VFDR-13-035 – 1NC VA0031B (Pressurizer PORV Isolation Valve), AP-45, Plant Fire, closes this valve.
- VFDR-04-106 – 1NC VA0033A (Pressurizer PORV Isolation Valve), AP-45, Plant Fire, closes valve
- VFDR-13-005 – 1CA VA0002 (Unit 1 Auxiliary Feedwater Pumps Suction from Auxiliary Feedwater Storage Tank Isolation Valve), AP-24, Loss of Plant Control Due to Fire or Sabotage, manually closes this valve when condensate storage tank level is low.

SSA RAI 05

LAR Section 4.2.1.1, "Comparison to NEI 00-01 Revision 2," states that post fire manual operation of rising stem valves in the fire area of concern, noted as an additional NEI 00-01 Revision 2 element, will be evaluated as part of the feasibility evaluation conducted as documented in "NFPA-805 Recovery Action Feasibility Review". LAR Attachment B, Table B-2, Section 3.2.1.2, identifies MCC-1435.00-00-0045 Rev. 0 - "NFPA 805 Transition Recovery Action Feasibility Review," as the referenced documentation. However, there is no identification of this element in the recovery action feasibility review. It appears that neither the assumptions nor the criteria in the recovery action feasibility review address this element.

Provide more detail with regard to which recovery actions require operation of rising stem valves in the fire area of concern. Identify where the criterion used in the evaluation is specifically identified, and how the criterion is evaluated.

Duke Energy Response:

MCC-1435.00-00-0045, Rev. 0, "NFPA 805 Transition Recovery Action Feasibility Review," did not identify valve operability issues of this kind since there were no rising stem valves identified at the time. However, Motor Operated Valves (MOVs) 1/2CA161C and 1/2CA162C, alternate Auxiliary Feedwater suction source, have subsequently been identified as rising stem valves. These valves would require opening prior to 18 hours after the initiating fire to provide an alternate suction water source for the Turbine Driven Auxiliary Feedwater (TDAFW) pump during Standby Shutdown System (SSS) operation.

The potential reliance on local operation of these valves following potential fire damage is being removed by plant modification. Engineering Changes (ECs) will be performed to install manual bypass butterfly valves around the valves of concern, which will eliminate the rising stem valve issue with 1/2CA161C and 1/2CA162C. MNS will revise the NFPA 805 LAR Attachment S, Table S-2, to commit to EC 109071 (Unit 1) and EC 109072 (Unit 2 - 2CA161C and 2CA162C only).

To ensure no future omission of justifying operation of rising stem valves, MCC-1435.00-00-0045 has been revised to include a requirement for an engineering evaluation for any rising stem valves added to the program that require a recovery action in the FA of concern.

Fire Modeling (FM) RAIs - 90-Day Responses

FM RAI 01.k

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:

k. Regarding the fires in the proximity of a wall or a corner, explain how the GFMTs approach was applied for a fire against a wall or in a corner. Explain how wall and corner effects in the ZOI and HGL timing calculations were accounted for, or provide a technical justification if these effects were not considered.

Duke Energy Response:

k. The following methodology was used for wall and corner effects in the Zone of Influence (ZOI) evaluation:

   Regarding separation distances, Calculation DPC-1535.00-00-0024, Rev. 0, "Generic Fire Modeling Treatments (GFMT)", Section 3.3.7 [Guidance for Fuel Packages Positioned in a Corner and Wall] states:

   1. If the fuel package is within 0.6 m (2 ft.) of a wall, then double the heat release rate and assume that the fire is centered at the fuel package edge adjacent to the wall.
   2. If the fuel package is within 0.6 m (2 ft.) of a corner, then quadruple the heat release rate and assume that the fire is centered at the fuel package corner nearest the wall corner.

   This GFMT is reflected in the MNS Fire Scenario Report, MCC-1535.00-00-0104, Rev. 3, Section 9.3 [Location Factor]. This section states:

   > The location of an ignition source relative to a wall or a corner may impact the zone of influence [ZOI]. While an ignition source walk down did not identify any fixed ignition sources as being located in a corner, ...Inverters... in the Battery Room were confirmed to be located against a wall. The inverters, which stand nearly 8 feet tall, were located against individual battery room walls that are approximately 8 feet high with significant free space between the top of the cabinet and the adjacent room and the overall battery room ceiling; therefore, the location of these cabinets against the wall has minimal impact on the heat release rate. The impact of the location on the zone of influence for these fixed ignition sources (all of which were equipped with a top mounted deflector shield) has been addressed in the scope of assumed target damage. Similarly for transients, if the postulated transient location ....was along a wall or in a corner, the zone of influence was adjusted accordingly.

Wall and corner effects were not applied to the HGL screening analysis. HGL effects were calculated for situations with and without the presence of localized fire exposures. In cases with localized fire exposures, where HGL formation was sufficient to impact the ZOI of the localized fire, a reduced critical heat flux value was used to determine whether or not a target would be damaged. The method used is described in the Generic Fire Modeling Treatments calculation, Section 6.1.2 [Combined Hot Gas Layer – Localized Fire Exposure Effects].

Because the overall heat input to the room is not increased by placement near a wall or corner, in order to address the initial change in rate, the room volumes (and ventilation parameters) would also be doubled and quadrupled, accordingly. The net impact on the room burnout calculation is, therefore, considered negligible.

FM RAI 02.e

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:

e. Explain how the damage thresholds for non-cable components (i.e., pumps, valves, electrical cabinets, etc.) were determined. Identify any non-cable components that were assigned damage thresholds different from those for thermoset and thermoplastic cables, and provide a technical justification for these damage thresholds.

Duke Energy Response:

e. In accordance with Appendix H.2 of NUREG/CR-6850, for major components such as motors, valves, etc., the fire vulnerability was assumed to be limited by the vulnerability of the power, control, and/or instrument cables supporting the component. As stated in the scenario development calculation, "In some fire scenarios, the target set may include another cabinet in which case application of the cable damage threshold would tend to be conservative since no credit would be taken for the protective nature of the enclosure." In other words, electrical cabinets are subject to the same damage thresholds as the cables in the analysis. No other non-cable components were assigned a damage threshold different from that which was used for cables.

FM RAI 02.f

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:

f. Describe the damage criteria that were used for exposed temperature-sensitive electronic equipment. Explain how temperature-sensitive equipment inside an enclosure was treated, and provide a technical justification for these damage criteria.

Duke Energy Response:

f. The sensitive electronics treatment at MNS is consistent with many aspects of the Fire PRA FAQ 13-0004. For example, the damage criteria used for temperature-sensitive electronic equipment inside of electrical cabinets was the same as that for thermoset cables. However, MNS has yet to officially incorporate FAQ 13-0004 since it was not approved when the Fire PRA was developed. The current sensitive electronics treatment in the MNS Fire PRA does not fully address the caveats in Fire PRA FAQ 13-0004 regarding sensitive electronics mounted on the surface of cabinets and the presence of louvers or vents. These caveats will be addressed in further detail in the response to MNS PRA RAI 16 (120 day response).

Probabilistic Risk Assessment (PRA) RAIs - 90-Day Responses

PRA RAI 01.d

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:

d) FSS-07: The disposition to the peer review assessment for FSS-07 (LAR Table V-2) does not specifically address how automatic suppression was credited other than in the MCA. Section 10.3 of the Fire Scenario Report identifies that the Halon suppression system was credited in the evaluation of turbine-driven auxiliary feedwater pump fire scenarios and that the generic unavailability for Halon systems from NUREG/CR-6850 was used in the evaluation of these scenarios. According to the PRA standard, the intent for CC-II is to "require a review of plant records to determine if the generic unavailability credit is consistent with actual system unavailability." Provide justification that the generic estimates for credited automatic suppression systems bound actual system unavailability based on an evaluation of plant records and that outlier behavior has not been experienced at MNS. If necessary, provide updated risk results as part of the aggregate change-in-risk analysis requested in PRA RAI 03 that appropriately accounts for actual automatic suppression system reliability/availability experience at MNS.

Duke Energy Response:

d) Fire PRA credit for automatic suppression is limited to the Halon systems in the Unit 1 and 2 Turbine Driven Auxiliary Feedwater Pump rooms. Credit was taken by applying a NUREG/CR-6850 generic unreliability factor of 0.05, which is reasonable given the system redundancy. Per the design basis specification for the Fire Protection systems, "There is one Auxiliary Feedwater Pump turbine Halon 1301 fire suppression system per unit incorporating one main Halon cylinder and one reserve Halon cylinder."

   A review of the MNS Fire Impairment Log records from February 2012 to September 2014 showed that for both Unit 1 and 2 Auxiliary Feedwater Pump turbine Halon 1301 fire suppression systems, unavailability was less than 0.05. Therefore, the non-suppression probability of 0.05 used in the MNS Fire PRA is appropriate for the limited number of scenarios which credit the Halon system, and updated risk results in the response to PRA RAI 03 are not necessary.

PRA RAI 12

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 the following information that is required to fully characterize the risk estimates.

LAR Section V.2.7 describes two MCR abandonment on loss-of-habitability scenarios, W1 and W2, where, in both cases, "failures were assumed which virtually eliminated all success paths other than the Standby Makeup Pump and the TDCA [turbine-driven auxiliary feedwater] pump from the SSF [Safe Shutdown Facility]." It is further explained that the conditional core damage probability (CCDP) for these scenarios is based on the highest CCDP for main control board (MCB) and non-MCB fires with additional failures as necessary to ensure no credit for functions that require continued presence in the MCR. Regarding this analysis, provide the following:

a) Summarize what "failures were assumed" and why they were assumed. Specifically, are they assumed because of general issues (e.g., unknown cable routing for functions always assumed failed) or are the assumptions only used for MCR abandonment scenarios?

b) An explanation of how the CCDPs account for the range of probabilities for properly shutting down the plant, and discussion of how they were applied in the scenario analysis. In doing so, provide examples over the full range of values utilized, a characterization of the scenarios to which these values are applied, and a summary of how each value is developed.

   This information should include explanations of how the following scenarios are addressed:

   i. Scenarios where the fire fails few functions aside from MCR habitability and successful shutdown is straightforward.

   ii. Scenarios where the fire could cause some recoverable functional failures or spurious operations that complicate the shutdown but successful shutdown is likely.

   iii. Scenarios where the fire induced failures cause great difficulty for shutdown by failing multiple functions and/or causing complex spurious operations that make successful shutdown unlikely.

c) Explanation of the timing considerations (i.e., total time available, time until cues are reached, manipulation time, and time for decision making) made to characterize scenarios in Part (b). Include in the explanation the basis for any assumptions made about timing.

d) Discussion of how the probability associated with failure to transfer control to the SSF is taken into account in Part (b).

e) Description of how the feasibility of the operator actions supporting the alternate shutdown pathway was considered by the scenario characterization performed in Part (b).

Duke Energy Response:

a. The MNS Fire PRA has the SSF and associated operator action to transfer control to the SSF fully incorporated into the PRA model. For control room abandonment, the SSF is the only success path credited in the Fire PRA. The primary functions of the SSF are reactor coolant pump seal cooling via the standby reactor coolant makeup pump and secondary side heat removal via the TDCA pump. To ensure that only SSF functions are credited as success paths for the Fire PRA control room abandonment scenarios, SSCs not failed by the control room fire were assumed to be failed or unavailable in the Fire PRA. Consequently, the failures that were referred to as "assumed" were not associated with assumed cable routing. These "assumed" failures were added as necessary to the abandonment scenarios to ensure no credit for functions that require continued presence in the MCR.

   Main Feedwater is an example of a failure added to the abandonment scenarios. Control of Main Feedwater is not available from the SSF. If the fire event requiring the SSF has not defeated Main Feedwater, transfer of control to the SSF will defeat Main Feedwater. To ensure sole reliance on the SSF, available credit for Main Feedwater recovery was removed. This removal was the nature of the "assumed" failures discussed in LAR section V.2.7.

b. As alluded to in the PRA RAI 12b question, the risk for the MCR abandonment scenarios should account for the impact of fire induced failures on the ability to shut down from an alternate location (SSF). The Fire PRA analysis was developed in a manner that addresses these impacts.

   Each MCR abandonment scenario encompasses the range of results from few functional failures to multiple functional failures, each condition (b.i, b.ii, & b.iii) leading to the most severe end state where the SSF is the sole remaining success path after abandonment. In the MNS Fire PRA, for the abandonment scenarios, the number of fire induced failures and spurious operations is based on the panel of origin that produces the highest conditional core damage probability (CCDP). Therefore, the abandonment scenarios account for the worst case impacts on the SSF regardless of a potentially more favorable outcome.

   The underlining assumption is that in the abandonment scenarios, any of the MCR fires could lead to conditions that require abandonment.

c. The scenarios addressed in part "b" of this RAI are modeled such that sequences rely on the SSF as the sole success path. Timing considerations for the PRA basic event that represents this success path are discussed below:

   Although the modeled abandonment scenarios are applicable to habitability, the human reliability analysis (HRA) timing takes into consideration that the operators will go to the SSF either for loss of habitability or for loss of function. Loss of function is the more limiting time available and therefore was the value that was used in the HRA. The analysis assumes that a loss of alternating current (AC) power which results in a loss of reactor coolant pump seal cooling occurs at the time the function is lost. This provides the smallest available timing used for the HRA probability.

   The timing values are based on the more limiting time available associated with a loss of function. The total time available from the loss of seal cooling to when a seal LOCA might result is estimated to be 13 minutes [Ref WCAP-15603, Rev.1] with a time of 7.5 minutes for cognition and recovery. The analysis assumes that the cue occurs at the beginning of the accident scenario (time 0). The timing does not explicitly address the Auxiliary Feedwater function since the seal cooling timing is the most limiting and there

is steam generator level instrumentation available upon transfer to the SSF that will auto-start the Turbine Driven Auxiliary Feedwater pump if necessary.

The discussion on timing, above, is applicable to the quantification results for the SSF human failure event, prepared for a future PRA update. The failure probability used in the MNS LAR submittal is considerably higher and therefore bounds the timing considerations provided above.

d. As discussed above, the SSF and the SSF human failure event are directly modeled in the Fire PRA; therefore, the "probability associated with failure to transfer control to the SSF" and the other random SSF equipment failures are directly quantified for each applicable scenario using the worst case HRA timing as discussed in "c" above.

e. Feasibility was considered in the development of the SSF human failure event failure probability. Time available for cognition and implementation of the actions are all taken into account for feasibility. This event is part of the Time Critical Operator action program and is trained upon with a completion time requirement of 10 minutes, which is shorter than the total time available (see "c"). Successful drills records indicate that the action is sufficiently feasible from a timing standpoint. The HRA analysis also considers the impact of lighting, environment, accessibility, and other performance shaping factors on feasibility. Regarding accessibility and environment, fires impacting the SSF fire area do not credit the SSF for recovery. Additionally, to address fires not impacting the SSF fire area, the SSF is located in the yard, and there are multiple pathways available for the operators to reach the SSF and avoid any fire effects.

PRA RAI 17

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 licensee's analysis indicates that the ZOI associated with a 142 kilo-watt (kW) heat release rate (HRR) (75th percentile) transient fire was used in almost all fires areas. Discuss the key factors used to justify the reduced rate below 317 kW per the guidance provided in the June 21,2012, memo from Joseph Giitter to Biff Bradley ("Recent Fire PRA Methods review Panel Decisions and EPRI 1022993, 'Evaluation of Peak Heat Release Rates in Electrical Cabinets Fires'," ADAMS Accession No. ML121171A583). Include in this discussion:

a) Identification of all fire compartments/areas where a ZOI for a reduced HRR of 142 kW (75th percentile) was used. The guidance in the referenced June 21, 2012,. memo indicates that a reduced HRR would be an exception supported by rigorous controls and restrictions. Please discuss how using a reduced HRR for almost all fire areas, if correct, is consistent with the guidance.

b) For each location (or group of similar locations) where a reduced HRR is credited, a description of the administrative controls that justify the reduced HRR including how location-specific attributes and considerations are addressed.

c) The results of a review of records related to violations of the transient combustible and hot work controls.

d) Confirm that 142 kW and 317 kW HRRs were the only transient fire sizes used in the FPRA.

Duke Energy Response:

a. A 142 kW fire size for general transient fires was applied in 18 out of the 40 fire compartments at MNS. These 18 compartments are part of the Auxiliary Building and are listed in the table below. The remaining majority of the site's fire compartments were locations where at least a 317 kW fire was assumed for general transient scenario development or the scenario development for the entire compartment was not refined beyond full room burnout.

FIRE COMPARTMENT	FIRE COMPARTMENT DESCRIPTION
1	Auxiliary Building Common EL 695' & Pipe Chase
2	Unit 1 Motor Auxiliary Feedwater (CA) Pump Room
2A	Unit 1 Turbine Driven CA Pump Room
3	Unit 2 Motor CA Pump Room
3A	Unit 2 Turbine Driven CA Pump Room
4	Auxiliary Building Common El 716'
13	Battery Rooms Common
14	Auxiliary Building Common El 733'
17A	Unit 1 Train A Switchgear HVAC Room
18A	Unit 2 Train A Switchgear HVAC Room
19	Unit 1 Cable Room
20	Unit 2 Cable Room
21	Auxiliary Building Common El 750'
25	Auxiliary Building Common El 767'
9-11	Unit 1 Train B Switchgear/Pen Room
10-12	Unit 2 Train B Switchgear/Pen Room
15-17	Unit 1 Train A Switchgear/Pen Room
16-18	Unit 2 Train A Switchgear/Pen Room

The June 21, 2012, memo mentioned in the RAI references the September 27, 2011, letter from Nuclear Energy Institute (NEI) to NRC (Subject: Recent Fire PRA Methods Review Panel Decisions: Clarification for Transient Fires and Alignment Factor for Pump Oil Fires), which includes an NRC accepted clarification of the guidance in NUREG/CR-6850 for selecting HRRs to be assumed for transient fires. The clarification acknowledges the HRR of 317 kW is a screening value that can be used throughout the analysis for a given plant unless there is reason to believe larger transient fires can be reasonably expected for a specific plant location based on the typical activities that occur. The clarification further states a lower screening HRR can be used for individual plant specific locations if the 317 kW value is judged to be unrealistic given the specific attributes and considerations applicable to that location. Locations within the plant under more rigorous controls or that have greater restrictions with respect to the introduction, handling, and placement of combustibles and/or the performance of hot work would be expected to have a lower HRR applied as compared to locations that have less rigorous controls and/or

restrictions. Further justification for how the guidance outlined in the September 27, 2011, letter is addressed in responses to parts "b" and "c" of this RAI.

b. The memo from Joseph Giitter to Biff Bradley ("Recent Fire PRA Methods review Panel Decisions and EPRI TR-1022993, 'Evaluation of Peak Heat Release Rates in Electrical Cabinets Fires', "ADAMS Accession No. ML 12171A583) endorses methods from the September 27, 2011, NEI submitted letter on "Recent Fire PRA Methods Review Panel Decision: Clarification for Transient Fires and Alignment Factor for Pump Oil Fires" with minor clarifications. Within this document, the following statements support the use of the lower heat release rate (HRR) of 142 kW:

- "A screening HRR of 317 kW can be used throughout the analysis for a given plant unless there is reason to believe that larger transient fires can be reasonably expected for a specific plant location based on the typical activities that occur. Conversely, a lower screening HRR can be used for individual plant specific locations if the 317 kW value is judged to be unrealistic given the specific attributes and considerations applicable to that location."
- "Locations within the plant that are under more rigorous controls or that have greater restrictions with respect to the introduction, handling, and placement of combustibles and/or the performance of hot work would be expected to have a lower HRR applied as compared to locations that have less rigorous controls and/or restrictions."

The following table provides dispositions relative to the recommendations bulleted above for each of the compartments listed in Part A of the RAI response where the 142 kW HRR fire size is credited.

FIRE COMPARTMENT	FIRE COMPARTMENT DESCRIPTION	Note
1	Auxiliary Building Common EL 695' & Pipe Chase	1
2	Unit 1 Motor CA Pump Room	2
2A	Unit 1 Turbine Driven CA Pump Room	2
3	Unit 2 Motor CA Pump Room	2
3A	Unit 2 Turbine Driven CA Pump Room	2
4	Auxiliary Building Common El 716'	3
13	Battery Rooms Common	4, 6
14	Auxiliary Building Common El 733'	3
17A	Unit 1 Train A Switchgear HVAC Room	5
18A	Unit 2 Train A Switchgear HVAC Room	5
19	Unit 1 Cable Room	2, 6
20	Unit 2 Cable Room	2, 6
21	Auxiliary Building Common El 750'	3
25	Auxiliary Building Common El 767'	3
9-11	Unit 1 Train B Switchgear/Pen Room	3, 6
10-12	Unit 2 Train B Switchgear/Pen Room	3, 6
15-17	Unit 1 Train A Switchgear/Pen Room	3, 6
16-18	Unit 2 Train A Switchgear/Pen Room	3, 6

Notes:

1. This compartment will be re-analyzed assuming the larger HRR of 317 kW and will no longer credit the 142 kW HRR.
2. The lower HRR of 142 kW is justified because the transient combustible control program has designated this compartment as needing additional controls beyond the basic requirements of the procedure (to be implemented during transition to NFPA 805). Per the procedure, administrative controls will be applied with regards to amount, duration and monitoring of combustible materials brought into these compartments based on a tiered approach which meet Chapter 3 requirements of NFPA 805 while at the same time incorporating the risk insights used in the PRA. There are limitations on materials being left unattended during work activities in these compartments which could require additional compensatory measures.
3. This compartment will be re-analyzed assuming the larger HRR of 317 kW in order to determine specific sections within these locations that require additional restrictions and support of the 142 kW HRR. MNS will update the transient combustible controls procedure to ensure that the sections identified require additional controls. A new implementation item will be created and submitted with the updated LAR to capture the areas that fall under this note.
4. Room 701 in Fire Compartment 13 is designated as a room with additional controls beyond the basic requirements of the procedure. Administrative controls are applied with regard to amount, duration and monitoring of combustible materials brought into these compartments based on a tiered approach which meet Chapter 3 requirements of NFPA 805 while at the same time incorporating the risk insights used in the PRA. This Fire Compartment also includes rooms 706 through 711 which were treated in a manner similar to full room burnout therefore the requirement for restrictions necessary for the assumption of the lower 142 kW fire size is not necessary. Also included in this Fire Compartment is room 648 which is the cable shaft. This location is not likely to accumulate combustible material due to its size.
5. This compartment is not designated as having additional controls beyond the basic requirements of the transient control procedure. However, the size and geometry of these rooms effectively restricts the size of the fuel package that can be practicably stored such that the 142 kW HRR is applicable.
6. This compartment predominantly contains electrical equipment and therefore significant combustibles normally associated with mechanical maintenance activities are not likely to accumulate in this area.

For the compartments listed above, introduction of transient fire loads is permitted only if conditions (i.e., separation distances and load limits) comply with the Transient Combustible Procedure (to be implemented during transition to NFPA 805).

Transient Combustible Procedure controls combustibles by the following methods:

- A tiered approach to required compensatory measures is established such that more stringent compensatory measures are required if transient combustible material is not separated from other transient combustible material to minimize quantity in one location.
- A tiered approach to required compensatory measures is established such that more stringent compensatory measures are required if transient combustible material is not separated from plant equipment susceptible to fire damage.

- At the end of each shift, housekeeping zones and work area owners should ensure all unnecessary materials and waste/trash material have been removed and properly disposed.

In addition to the Transient Combustible Procedure, other Administrative controls that justify the reduced HRR for all the compartments listed above include:

- The Auxiliary Building is a radiation protection area, therefore the allowance of any combustibles are strictly limited. Site personnel are directed to reduce the amount of combustibles (if any) per radiation worker training.
- Per the Housekeeping procedure, the Auxiliary Building requires good general housekeeping practices, such as ensuring that no accumulation of dirt, dust, trash, or improperly stored equipment.

Furthermore, observations from applicable test data were considered in support of the use of the 142 kW transient HRR.

- The materials composing the fuel packages included in Table G-7 of NUREG/CR-6850 (e.g., eucalyptus duff, one quart of acetone, 5.9 kg of methyl alcohol, etc.) are not representative of the typical materials expected to be located in the auxiliary building fire compartments.
- A review of the transient ignition source tests in Table G-7 of NUREG/CR-6850 indicates that of the type of transient fires that can be expected in these rooms (i.e., polyethylene trash can or bucket containing rags and paper) were measured at peak HRRs of 50 kW or below.
- NUREG/CR-4680, "Heat and Mass Release for Some Transient Fuel Source Fires: A Test Report", dated October 1986, documents a series nine trash fire characterization tests. Five different fuel packages made up of small to moderate trash accumulations were ignited to record information on heat and mass release rate properties of fires in fuel packages of this type. The results noted that none of the peak HRRs exceeded 150 kW. Typical HRRs were in the range of 20-50 kW.

In summary, for a majority of the fire compartments discussed in the table above, application of the higher 317 kW HRR may yield unrealistic results given the rigorous administrative controls outlined in the Transient Combustible Procedure to be implemented during NFPA 805. The Transient Combustible Procedure will be an administrative procedure with a tiered approach for meeting the fire prevention requirements of NFPA 805 Chapter 3 while at the same time incorporating risk insights used in the PRA. This procedure, coupled with the training provided to station personnel, affirms that utilizing a reduced HRR of 142 kW is appropriate for use in the Fire PRA at MNS. The remaining fire compartments either credit the room geometry (see note 5 above) that will limit the fire size, or will be re-evaluated to determine if a 317 kW HRR is more appropriate and/or if additional controls are required. Any impact to the results will be evaluated as part of PRA-RAI-03.

c) A review of Problem Investigation Program (PIP) reports was conducted to identify violations of transient combustible and hot work controls between May 2012 and October 2014. There were forty-two transient combustible violations and only two minor hot work related violations. No further discussion will be made for the two minor hot work violations.

   Regarding the forty-two transient combustible related violations, nineteen are in the Turbine Building and Service Building fire areas where a higher heat release rate (317 kW) was applied. Consequently, these nineteen violations are not relevant to this RAI response. There were also three transient combustible related violations in FAs 26, 27, and SSF where an 'A' case scenario (room burnout scenarios where everything routed in

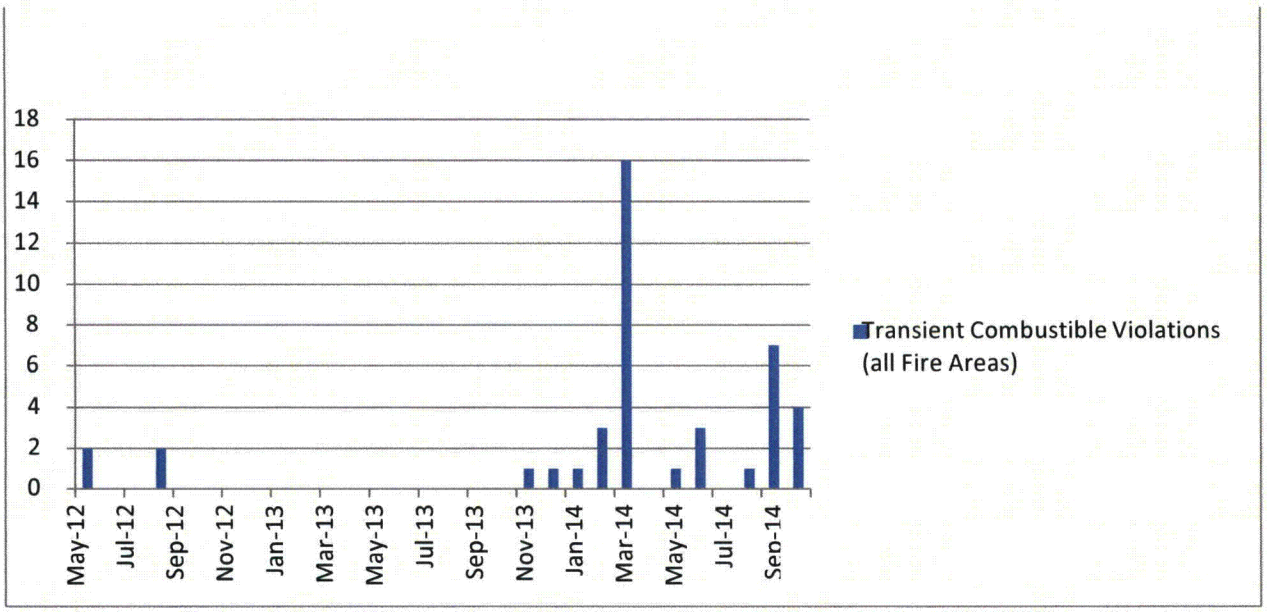
the room is assumed to be fire damaged) was utilized. These three are also excluded from further evaluation since fire size is not relevant for these room burnout cases. One violation did not take place in a designated fire area so it was excluded from further consideration.

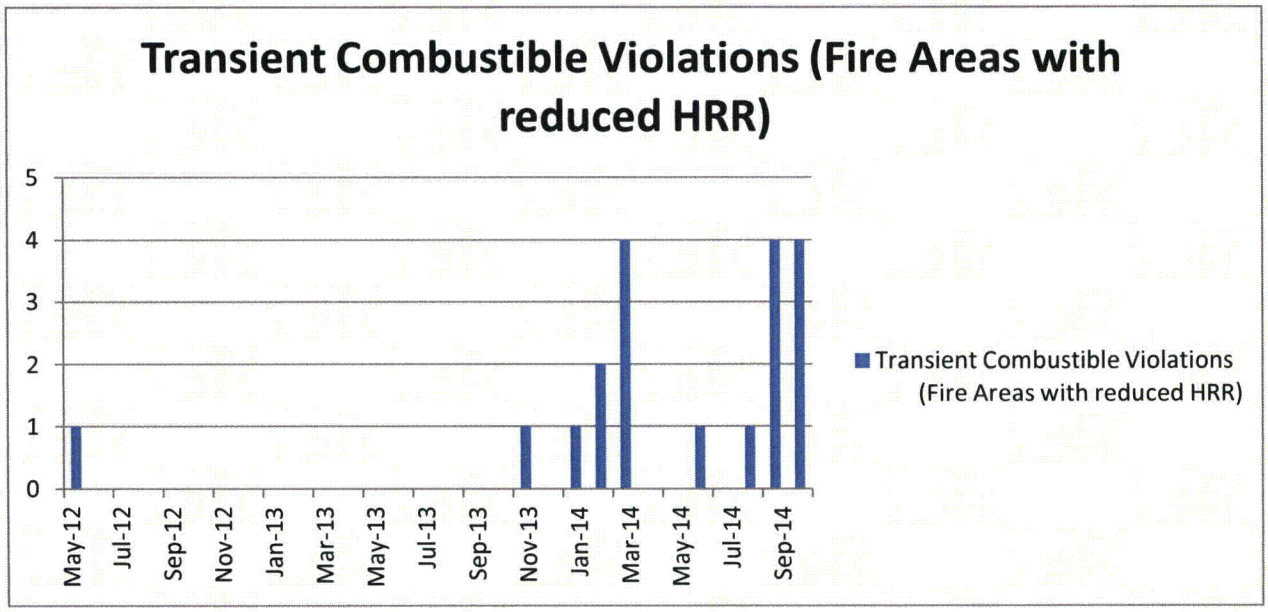
The remaining nineteen transient combustible related violations identified in areas associated with a reduced HRR of 142 kW were identified in FAs 2, 3, 3A, 14, 19, 20, 21, and 25. The violations were primarily administrative in nature, such as missing documentation needed to exceed the limit for combustible material in the fire area.

The site is currently under a Transient Combustible Improvement Action plan, which has yielded a proactive fire program and heightened sense of awareness over time as more violations were identified during the transition from Appendix R to NFPA 805. This is evident in the increase of violations that took place in March 2014. A second increase of violations appeared in September 2014. However, this can be attributed to the Unit 1 outage (outage start date was September 13, 2014). Although four violations took place in FA 14 during this month, this is not an adverse trend considering how large FA 14 is and the fact the violations were not severe. One violation was for a slightly overfilled 55 gallon drum and the other three were administrative in nature (missing documentation for a Nuclear Site Directive [NSD] 313-1 form which documents the evaluation done by the site Fire Protection Engineer). The four violations identified in October 2014 (2 in FA 14 and 2 in FA 25) were also administrative violations as they were missing documentation for an NSD 313-1 evaluation. Additionally, the violations identified during the outage do not apply to an at-power Fire PRA.

The goal of the Transient Combustible Improvement Action plan is to increase station awareness concerning transient combustibles. Consequently, these violations are expected to diminish. As the sensitivity for transient combustibles is increased, a positive change in fire safety culture at the station has taken place. Accordingly, the MNS Fire PRA reflects the plant as designed and operated with respect to the Transient Combustible program.

Transient Combustible Violations (all Fire Areas)





d) Only the 142 kW and 317 kW HRRs were used in the Fire PRA. In some cases (e.g. room burnout), the transient HRR was immaterial to the quantification.