Entergy

Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802
Tel 479-858-3110

Jeremy G. Browning
Vice President - Operations
Arkansas Nuclear One

1CAN081501

August 12, 2015

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

SUBJECT: 120-Day Response to Request for Additional Information
Adoption of National Fire Protection Association Standard NFPA-805
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

Dear Sir or Madam:

By letter dated May 5, 2015 (Reference 2), the NRC requested additional information associated with the Entergy Operations, Inc. (Entergy) request to amend the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TS) and licensing bases to comply with the requirements in 10 CFR 50.48(a), 10 CFR 50.48(c), and the guidance in Regulatory Guide (RG) 1.205, “Risk-Informed Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants.” The amendment request followed Nuclear Energy Institute (NEI) 04-02, “Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c).” This submittal described the methodology used to demonstrate compliance with, and transition to, National Fire Protection Association (NFPA) 805, and included regulatory evaluations, probabilistic risk assessment (PRA), change evaluations, proposed modifications for non-compliances, and supporting attachments.

Based on the complexity of the questions included in the Reference 2 letter, the NRC established response due-dates of 30, 60, 90, or 120 days, from the date of the ANO-1 NFPA 805 Audit Exit Meeting, April 23, 2015. Responses to the 30-day and 60-day RAIs were included in References 3 and 4, respectively. One 60-day RAI (SSA RAI 02) was included in the 30-day response and two 90-day RAIs (FM RAI 03 and PRA RAI 02.c) were included in the 60-day response. All remaining 90-day RAI responses were included in Reference 5. Enclosed are responses to all 120-day RAIs.

Changes or additional information, as detailed in this letter, with respect to the original Entergy request (Reference 1) have been reviewed and Entergy has determined that the changes do not invalidate the no significant hazards consideration included in the Reference 1 letter.

In accordance with 10 CFR 50.91(b)(1), a copy of this application is being provided to the designated Arkansas state official.

No new commitments have been identified in this letter.

If you have any questions or require additional information, please contact Stephenie Pyle at 479-858-4704.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on August 12, 2015.

Sincerely,

ORIGINAL SIGNED BY JEREMY G. BROWNING

JGB/dbb

Attachment: 120-day Responses to Request for Additional Information – ANO-1 Transition to NFPA-805

REFERENCES:

1. Entergy letter dated January 29, 2014, *License Amendment Request to Adopt NFPA-805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)* (1CAN011401) (ML14029A438)

2. NRC letter dated May 5, 2015, *Arkansas Nuclear One, Unit 1 – Request for Additional Information Regarding License Amendment Request to Adopt National Fire Protection Association Standard 805* (TAC No. MF3419) (1CNA051501) (ML15091A431)

3. Entergy letter dated May 19, 2015, *Response to Request for Additional Information – Adoption of National Fire Protection Association Standard NFPA-805* (1CAN051501) (ML15139A196)

4. Entergy letter dated June 16, 2015, *60-Day Response to Request for Additional Information – Adoption of National Fire Protection Association Standard NFPA-805* (1CAN061501) (ML15167A503)

5. Entergy letter dated July 21, 2015, *90-Day Response to Request for Additional Information – Adoption of National Fire Protection Association Standard NFPA-805* (1CAN071501) (ML15203A205)

cc: Mr. Marc L. Dapas
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
1600 East Lamar Boulevard
Arlington, TX 76011-4511

NRC Senior Resident Inspector
Arkansas Nuclear One
P. O. Box 310
London, AR 72847

U. S. Nuclear Regulatory Commission
Attn: Ms. Andrea E. George
MS O-8B1
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Mr. Bernard R. Bevill
Arkansas Department of Health
Radiation Control Section
4815 West Markham Street
Slot #30
Little Rock, AR 72205

Attachment 1 to

1CAN081501

120-day Responses to Request for Additional Information
ANO-1 Transition to NFPA-805

120-DAY RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
ANO-1 Transition to NFPA-805

By letter dated May 5, 2015 (Reference 2), the NRC requested additional information associated with the Entergy Operations, Inc. (Entergy) request (Reference 1) to transition the Arkansas Nuclear One, Unit 1 (ANO-1), fire protection licensing basis to National Fire Protection Association (NFPA) Standard NFPA 805, *Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)*. Included below are Entergy responses to all questions requiring a 120-day response with respect to the *request for additional information* (RAI) (Reference 2). In accordance with Reference 2, final risk quantifications will be submitted following NRC acceptance of all RAI responses.

Safe Shutdown Analysis (SSA)

<u>SSA RAI 11</u>

NFPA 805 Section 2.4.2.2.1 states that circuits required for the nuclear safety functions shall consider fire-induced failure modes such as hot shorts (external and internal), open circuits, and shorts to ground.

In LAR Attachment C and Attachment S, Table S-1, "Plant Modifications," the licensee identified modifications on valves (e.g., CV-1221, CV-1405, CV-1406, SG-1, SG-2, SG-3, and SG-5) that will be credited to meet the requirements of IN 92-18, and stated that the modification will add an "inhibit" circuit which will preclude spurious opening of the MOV due to intercable or intracable hot shorts.

a) Please clarify if these modifications are credited to meet deterministic approach of NFPA 805 Section 4.2.3

b) Please describe the details of the modification, including the details of the "inhibit" circuit.

c) For each fire area that credits these modifications, please discuss how the effects of hot shorts (external and internal), open circuits, and shorts to ground were evaluated to ensure that the desired nuclear safety function of the valves will not adversely affect the ability to achieve the nuclear safety performance criteria credited in the fire areas that credit these modifications. As part of this discussion, address the potential for "collateral damage" as a result of the energetic failure of other circuits in the same raceway/wireway as the shorting portion of the circuit. In NUREG/CR-7100, "Direct Current Electrical Shorting in Response to Exposure Fire (DESIREE- Fire): Test Results," Final Report, April 2012 (ADAMS Accession No. ML 121460107), an NRC-sponsored fire test, it was concluded that open circuits in adjacent cables resulting from energetic faulting will not occur as long as the fusing used is 10 amps or less. For circuits fused greater than this, testing showed that the fuses may not clear on a fault, thereby causing the dc cable to open circuit, resulting in repeated arcing of conductors. This repeated arcing could result in an open circuit in an adjacent cable.

d) During the NFPA 805 audit performed the week of April 20-23, 2015, the licensee described that a new "shorting switch" will be installed in the control room as part of this modification. Please describe how fire damage to the new "shorting switch" will not affect the desired nuclear safety function required to achieve the nuclear safety performance criteria.

Response

ANO-1 modifications identified in LAR Attachment S, Table S-1, "Plant Modifications," that use an inhibit circuit are listed in the following three tables. Additional data shown in the tables was obtained from electrical schematics, the ANO-1 fuse list (E-1020 drawing series), and the Plant Data Management System (PDMS), which is the circuit and raceway database.

Large Early Release Frequency (LERF) Reduction

(Credited in Fire Area G – Control Room)

Valve	Table S-1	Type	Voltage	Fuse (AMPS)	Wire Size (AWG)	1-Way Length of Inhibit (ft)	Notes
CV-1052	S1-10	AOV	125 VDC	15	#14	328	IN SERIES WITH CV-1053
CV-1053	S1-11	MOV	120 VAC	1.8	#14	207	IN SERIES WITH CV-1052
CV-4400	S1-16	AOV	125 VDC	15	#14	306	IN SERIES WITH CV-4446
CV-4446	S1-17	MOV	120 VAC	2.25	#14	311	IN SERIES WITH CV-4400
CV-5611	S1-18	MOV	120 VAC	1.8	#14	181	IN SERIES WITH CV-5612
CV-5612	S1-19	MOV	120 VAC	2.25	#14	464	IN SERIES WITH CV-5611
CV-7401	S1-20	AOV	125 VDC	15	#14	475	IN SERIES WITH CV-7403
CV-7402	S1-21	AOV	125 VDC	15	#14	475	IN SERIES WITH CV-7404
CV-7403	S1-22	AOV	125 VDC	15	#14	667	IN SERIES WITH CV-7401
CV-7404	S1-23	AOV	125 VDC	15	#14	642	IN SERIES WITH CV-7402

MOV = Motor Operated Valve AOV = Air Operated Valve

Core Damage Frequency (CDF) Reduction

Valve	Table S-1	Type	Voltage	Fuse (AMPS)	Wire Size (AWG)	1-Way Length of Inhibit (ft)
CV-1221	S1-12	MOV	120 VAC	1.8	#14	157
CV-1405	S1-13	MOV	120 VAC	1.8	#14	214
CV-1406	S1-14	MOV	120 VAC	1.125	#14	181
SG-1	S1-27	MOV	120 VAC	1.8	#14	389
SG-2	S1-28	MOV	120 VAC	1.8	#14	187
SG-3	S1-29	MOV	120 VAC	1.8	#14	295
SG-4	S1-30	MOV	120 VAC	2	#14	177

Non-Power Operational (NPO) Modes

Valve	Table S-1	Type	Voltage	Fuse (AMPS)	Wire Size (AWG)	1-Way Length of Inhibit (ft)
CV-1404	S1-35	MOV	120 VAC	.6	#14	93

a) No credit is taken for modifications identified as CDF reduction in areas that use the deterministic approach of NFPA 805, Section 4.2.3. LERF reduction is only considered in the Fire PRA (FPRA) and NPO is a special case where pinch points can be qualitatively assessed based upon risk.

b) The ANO-1 inhibit circuit uses conductors and a control switch to intentionally short across the target coil when the device is in its normal de-energized state. This is done so that the coil cannot be electrically actuated by fire-induced failures where a source conductor applies a sufficient voltage either from intracable or intercable faults. A normally closed contact on the same control switch used for controlling the device is used in the inhibit circuit. Manipulation of this “break-before-make” manual control switch by an Operator will break the shorting connection, allowing the device to be actuated.

c) The inhibit circuit is effectively a low resistance jumper across the coil, which functions to prevent application of a sufficient voltage to actuate the device (pickup voltage). If the fault current through the inhibit circuit is sufficiently high, the protective device will clear the fault and remove power. Where the current is not adequate due to overall resistance, the voltage drop across the low resistance inhibit circuit will be less than the pick-up voltage of the contactor. The cable of interest for the valves listed that can cause spurious operation is the cable between the Control Room handswitch and either the Motor Control Center (MCC) or solenoid coil.

Considering an intracable fault, the application of a voltage via a fault at the contactor will be bounding as this will yield the highest resistance for the inhibit circuit. Excluding modifications for LERF reduction, the longest one-way length is less than 400 feet or a total length of less than 800 feet. The resistance of #14 AWG wire is nominally 2.73 ohms/1000 feet as obtained from the ANO-1 Millstone study CALC-95-E-0001-06. The largest fuse of 2 amps is for sluice gate SG-4 and the largest control power transformer (CPT) is 150 VA with secondary resistance of 2.91 ohms (worst-case minimum resistance of Allis Chalmers 150 VA CPT), also obtained from the ANO-1 Millstone study. For a theoretical zero-resistance bolted fault the total fault current would be 21.3 amps (120 volts / [2.73 + 2.91] ohms) and 58 volts (21.3 amps * 2.73 ohms) across the inhibit circuit, which is the same as across the coil.

ANO-1 uses Allis Chalmers, ITE, and Siemens contactors. A review of acceptance testing data for representative Siemens contactors supplied with ANO purchase order PO-944963 shows the minimum pickup for a NEMA size 1 contractor is greater than 87 volts. Contactors used in EPRI, CAROLFIRE, and DESIREE- FIRE tests had pickup voltages that ranged from 72 volts to 93.9 volts (CAROLFIRE section 8.4.1 & DESIREE- Fire Table A-13). The fire tests and ANO acceptance tests support using 60% (72 V) of the rated coil voltage of 120 V as being the minimum voltage that will result in contactor pick-up. In the scenario described, a value less than 50% of the rated voltage, insufficient to cause contactor pickup, is briefly applied before the 2-amp fuse clears the circuit. Any resistance at the fault would only serve to reduce the current and the voltage across the coil accordingly. Figure 4.1 of NUREG/CR-6931, "Cable Response to Live Fire," indicates that intracable resistance between conductors will be greater than the combined resistive values of the inhibit circuit and CPT secondary.

The modified LERF valves that prevent an uncontrolled vent path to atmosphere have DC circuits in addition to AC, but are only credited in Fire Area G. There are two normally closed valves in series, controlled from separate panels, for each protected LERF path that would need to fail open due to fire-induced spurious operation. Limiting protection of these circuits to Fire Area G results in a smaller inhibit circuit resistance than used above for valves that are protected for the entire distance from the control room to the MCC. The evaluation for non-LERF valves bounds those modified for LERF.

An intercable fault could result in spurious operation if the target cable has not previously sustained intracable damage that would prevent functionality of the conductors required to energize the coil. Failure of the inhibit circuit from an open conductor will be necessary to facilitate spurious operation. The fault would also need to be of a sufficiently low resistance such that a voltage greater than the pickup voltage is applied across the coil. Individually, each of these conditions appear unlikely to occur, and as a whole make intercable spurious actuation non-credible, based upon fire tests performed for both AC and DC circuits. Supporting excerpts from each of these tests include:

EPRI TR-1003326 "Characterization of Fire-Induced Circuit Faults - Results of Cable Fire Testing"

Section 2.3.1 - *"Further, as testing shows, open circuits appear to be an extremely improbable initial failure mode for fire-induced cable failures (i.e., conductor insulation damage will result in a hot short or short-to-ground well before temperature reaches a level that will cause the conductor to melt).*

Section 14.4 – *"No open circuit faults occurred during the Test Program. Open circuits do not appear to be a credible primary cable failure mode for fire-induced cable faults."*

NUREG/CR-6931

Volume 1 Section 8.1.3: *"In the end, no single instance of a spurious actuation on any [Surrogate Circuit Diagnostic Units] SCDU configured to simulate an MOV control circuit was attributed to intercable interactions during any of the CAROLFIRE tests."*

Volume 1 Section 8.3.3: *"The IRMS [Insulation Resistance Measurement System] data from the CAROLFIRE tests have confirmed one key observation that was made during the NEI/EPRI tests; namely, intracable shorting is expected to be the predominant mode of initial cable failure rather than shorts to an external ground."*

NUREG/CR-7100

Section 6.5.1 *"Of the four intercable bundles tested, all indicated cable-to-cable interactions between the target cable and one or more of the energized source cables. This was not unexpected given the test configuration, which, as noted above, was designed to optimize the potential for such interactions. In Tests 45, 46, and 48 the intercable interactions occurred only after the target cable had failed internally."*

Section 6.5.3 *"The energizing voltage difference observed on the target cable reached a maximum of only about 15Vdc, indicating that the cable-to-cable resistance values, while degrading, remained quite high. The voltages imposed on the target conductors in this case were not sufficient to induce a spurious operation."*

With regard to the possibility of collateral damage, a review of NUREG/CR-7100 shows that in Penlight Test #3 (Thermoset) and Penlight Test #10 (Thermoplastic) there was destructive damage where opening (melting) of cable conductors was observed. Both tests involved medium voltage switchgear trip and close circuits consisting of #12 AWG conductors fused respectively at 35 and 15 amps. Penlight Test 3 (Thermoset) had significant arcing associated with the cables and some conductors of one cable failed open (melted). Penlight Test 10 (Thermoplastic) had failed conductors in both cables, but the applicability of this test is debatable as ANO uses thermoset cables. No statement or conclusion is made in NUREG/CR-7100 for either test that the cable damage is collateral rather than self-inflicted. Postulating collateral damage that could cause failure of the inhibit circuit followed by spurious actuation requires the following:

1. The target cable does not sustain intracable damage that would prevent functionality of the conductors required to energize the coil.
2. An adjacent DC powered cable undergoes the aggressive arcing described above without clearing of the fuse.
3. The collateral damage to the target cable is limited to the melting of the conductors required for the inhibit circuit exclusive of damaging conductors required to energize the coil.
4. An intercable fault of the correct voltage is applied from a cable that has not sustained a sufficient level of fire-induced damage as to render it non-functional.

Considered individually, each of these four items is plausible, but as a sequence of events, becomes a non-credible failure when considering the evidence from NUREG/CR-6931 (CAROLFIRE) and NUREG/CR-7100 (DESIREE- Fire) as described above for intercable faults.

d) Shorting switches for the ANO-1 modifications are General Electric (GE) CR-2940 handswitches. According to an environmental qualification (EQ) report for the GE CR2940 (ANO Vendor File V-29, Item 25), the handswitch body and internal plunger are made of a phenolic resin and the spring retaining washer is compressed asbestos, with the remaining parts being brass or steel. The normal and shelf state position for these switches are the same (“off” or “center”) with the spring pushing the internal plunger forward with a nominal 4.8 pounds of force (ANO Vendor File V-29, Item 25, Page 8). A single contact assembly bridges the Normally Closed (N/C) and Normally Open (N/O) contacts. This means that it is not physically possible to make-up the N/C and N/O contacts at the same time. Disassembly of a switch body shows the contact assembly would have to travel about 0.25” to make up the N/O set of contacts.

   Fire testing of various handswitches documented in Exelon Nuclear Evaluation EC-EVAL A1831999 01, “Evaluation of Shorting Switch Modification,” Rev 0, December, 2011, includes the GE CR-2940. This Exelon evaluation investigates the use of a "shorting switch" that can prevent the spurious energization of a particular coil (relay, solenoid, and contactor). National Technical System, Inc., performed the actual fire testing and the test report is included as an attachment to the Exelon report. A peak temperature of 575 °F (300 °C) was maintained for 15 minutes following a ramp-up from room temperature with a subsequent cool down with natural ventilation. The heat release rate selected to obtain this temperature was chosen based upon a review of NUREG/CR-4527 and consideration of cabinet orientation, qualified cables (IEEE 383), and higher in-situ fuel source. ANO-1 Control Room cabinets fall into this profile and the testing performed is appropriate. The GE CR-2940 switch performed well with two tested exceptions:

   - In one test (17D), the N/C contact opened, but no closure of the N/O contact occurred. In the ANO application this would have resulted in the opening of the inhibit circuit, but not caused spurious operation due to the switch.

- In one test (17E), an N/O contact closed without opening of the N/C contact. The suspected cause was flow of the panel-mounting gasket into the switch, creating an internal conductive path. In the ANO application, the inhibit circuit would have remained intact.

Tests 17D and 17E were “Horizontal” tests where the CR-2940 switch was mounted in a bench board rather than a vertical panel. The ANO applications have the switches mounted in vertical panels where gravity would not be acting to pull the contact assembly away from the N/C contact set or flow panel-mounting gasket into the switch.

The majority of internal cable bundles in ANO-1 Control Room vertical panels are vertically run on the sides. Interface of the field cables to the internal wiring is at terminal blocks also mounted to the sides. Smaller horizontal bundles of internal wiring break out from the vertical bundles and connect to handswitches and instruments mounted on the front. This places the handswitch out of the plume for any fire originating in the vertical cable bundles that should serve to limit the temperature the switch experiences for fires within the panel. Cables and internal wiring are thermoset and meet IEEE 383 requirements.

NUREG/CR-4527 scoping tests ST-6 through ST-9 are cabinet fire propagation tests on qualified cable. For each test, the use of an accelerant and combustible, that would not be present in ANO-1 Control Room panels, was utilized as an ignition source. As a result of the scoping tests, the authors reached a conclusion that fires of qualified cable in vertical cabinets do not spread throughout the cabinet and the thermal environment in the enclosure does not become severe enough to cause melting of components or result in flashover. This indicates that once a fire starts in a cable bundle, its effects will be localized, and any failures will occur at that location before the handswitch is damaged.

The internal configuration of flame retardant materials in ANO-1 Control Room panels makes a fire unlikely to propagate and then sustain itself for a sufficient period of time to damage the robustly designed GE CR-2940 handswitch. Failure of conductor insulation in either the jacketed cables or internal panel wiring is expected to occur prior to failure of the handswitch due to the heavier construction of the switch and its greater thermal mass. The orientation of the handswitch should prevent changes that would open the N/C contacts that occurred in test 17D of the Exelon report. In summary, postulated damage to the shorting switch would not be sufficient to affect the desired nuclear safety function required to achieve the nuclear safety performance criteria.

Probabilistic Risk Assessment (PRA)

Note: Responses to PRA RAIs 03 (without quantitative results), 10, 11, 13, 15, 17 (without quantitative results), and 18 are included below. In accordance with Reference 2, final risk quantifications will be submitted following NRC acceptance of all RAI responses.

<u>PRA RAI 03</u> – Integrated Analysis

Section 2.4.4.1 of NFPA-805 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, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the

Licensing Basis," Revision 2, dated May 2011 (ADAMS Accession No. ML 100910006), provides quantitative guidelines on CDF and 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.

The PRA methods currently under review in the LAR include:

- PRA RAI 1.a regarding spatial separation
- PRA RAI 1.b regarding fire barriers
- PRA RAI 1.d regarding fire propagation from electrical cabinets
- PRA RAI 1.h regarding circuit failure likelihood analysis
- PRA RAI 1.j regarding modeling new fire Human Error Events
- PRA RAI 1.k regarding state of knowledge correlation (SOKC)
- PRA RAI 2.a regarding impact of phenomenological conditions
- PRA RAI 2.d regarding counting operational demands
- PRA RAI 2.e regarding counting failures
- PRA RAI 4 regarding reduced transient HRRs
- PRA RAI 5 regarding treatment of sensitive electronics
- PRA RAI 7 regarding propagation of fire from >440 V electrical cabinets
- PRA RAI 8 regarding use of the transient frequency adjustment factors
- PRA RAI 9 regarding fire propagation in the MCR
- PRA RAI 11 regarding crediting MCR abandonment
- PRA RAI 12 regarding multiple versus single cables
- PRA RAI 14 regarding large reduction credit for modifications
- FM RAI 1.k regarding evaluation of MCR abandonment times

Please provide the following information:

a) Results of an aggregate analysis that provides the integrated impact on the fire risk (i.e., the total transition CDF, LERF, ΔCDF, ΔLERF) of replacing specific methods identified above with alternative methods which are acceptable to the NRC. In this aggregate analysis, for those cases where the individual issues have a synergistic impact on the results, a simultaneous analysis must be performed. For those cases where no synergy exists, a one-at-a-time analysis may be done. For those cases that have a negligible impact, a qualitative evaluation may be done. It should be noted that this list may expand depending on NRC's review of the responses to other RAIs in this document.

b) For each method (i.e., each bullet) above, please explain how the issue will be addressed in 1) the final aggregate analysis results provided in support of the LAR, and 2) the PRA that will be used at the beginning of the self-approval of post-transition changes. In addition, provide a method to ensure that all changes will be made, that a focused-scope peer review will be performed on changes that are PRA upgrades as defined in the PRA standard, and that any findings will be resolved before self-approval of post-transition changes.

c) In the response, explain how the RG 1.205 risk acceptance guidelines are satisfied for the aggregate analysis. If applicable, include a description of any new modifications or operator actions being credited to reduce delta risk as well as a discussion of the associated impacts to the fire protection program.

d) If any unacceptable methods or weaknesses will be retained in the PRA that will be used to estimate the change in risk of post-transition changes to support self-approval, explain how the quantification results for each future change will account for the use of these unacceptable methods or weaknesses.

Response

a) Final risk quantification will be provided after NRC concurrence is obtained with the approach described in the various RAI responses. This will ensure the integrated response reflects NRC accepted approaches.

b) The following table provides the requested disposition of each of the RAIs listed above:

RAI No. / Description	Disposition with respect to the final integrated analysis and the aggregated results provided in support of the LAR (to be incorporated into the final PRA RAI 03 response which includes the final Fire PRA quantification results)	Disposition with respect to the self-approval model for post-transition changes
PRA RAI 1.a regarding spatial separation	Spatial separation is not credited. The zone of influence is allowed to cross non-barrier boundaries. No change to the Fire PRA (FPRA) model is required.	Same approach as that used for the final integrated analysis provided in support of the LAR.
PRA RAI 1.b regarding fire barriers	There are no active fire protection systems supporting the Multi-Compartment Analysis (MCA) fire barriers that require an actuation system (i.e., barrier features which credit systems that require signals from cables or a detection system) as part of any physical analysis unit (PAU) boundary at ANO-1 (e.g., water curtains). The MCA is being revised to sum the generic barrier failure probabilities for each type of barrier present between PAUs.	Same approach as that used for the final integrated analysis provided in support of the LAR.

RAI No. / Description	Disposition with respect to the final integrated analysis and the aggregated results provided in support of the LAR (to be incorporated into the final PRA RAI 03 response which includes the final Fire PRA quantification results)	Disposition with respect to the self-approval model for post-transition changes
PRA RAI 1.d regarding fire propagation from electrical cabinets	The panel factor approach was eliminated prior to submitting the LAR. Severe and non-severe panel fires have been defined based on the zone of influence up to and excluding the nearest target. The methodology used is based on data provided in NUREG/CR-6850, Appendices E and H, and the methodology defined in the Generic Fire Modeling Treatments (GMFT).	Same approach as that used for the final integrated analysis provided in support of the LAR.
PRA RAI 1.h regarding circuit failure likelihood analysis	Circuit failure likelihood values used will be consistent with the values specified in NUREG/CR-7150.	Same approach as that used for the final integrated analysis provided in support of the LAR.
PRA RAI 1.j regarding modeling new fire Human Error Events	The FPRA will incorporate the update to the Human Reliability Analysis (HRA) methodology that is consistent with developing detailed human error probabilities (HEPs) as outlined in NUREG-1921.	Same approach as that used for the final integrated analysis provided in support of the LAR.
PRA RAI 1.k regarding state of knowledge correlation (SOKC)	A SOKC was applied to ignition frequencies, circuit failure probabilities, non-suppression probabilities, and HRA basic events. The SOKC will be addressed in the final FPRA documentation.	Same approach as that used for the final integrated analysis provided in support of the LAR.
PRA RAI 2.a regarding impact of phenomenological conditions	The FPRA model will be revised to address the phenomenological issues as identified in response to PRA RAI 2.a.	Same approach as that used for the final integrated analysis provided in support of the LAR.

RAI No. / Description	Disposition with respect to the final integrated analysis and the aggregated results provided in support of the LAR (to be incorporated into the final PRA RAI 03 response which includes the final Fire PRA quantification results)	Disposition with respect to the self-approval model for post-transition changes
PRA RAI 2.d regarding counting operational demands	The internal events model is not altered by the response to this RAI (see RAI response for basis). Therefore, the resolution of this RAI does not impact the PRA quantification.	Not applicable to the development of the post transition self-approval model, since this RAI did not impact the integrated analysis and the aggregated results provided in support of the LAR.
PRA RAI 2.e regarding counting failures	The internal events model is not altered by the response to this RAI (see RAI response for basis). Therefore, the resolution of this RAI does not impact the PRA quantification.	Not applicable to the development of the post transition self-approval model, since this RAI did not impact the integrated analysis and the aggregated results provided in support of the LAR.
PRA RAI 4 regarding reduced transient HRRs	Reduced heat release rate values are used in distinct areas with restricted transient controls in the new fire protection program.	Same approach as that used for the final integrated analysis provided in support of the LAR.
PRA RAI 5 regarding treatment of sensitive electronics	The impact of the results of walkdowns of sensitive electronics will be incorporated into the final FPRA quantification as necessary, consistent with the methodology outlined in FAQ 13-0004.	Same approach as that used for the final integrated analysis provided in support of the LAR.
PRA RAI 7 regarding propagation of fire from > 440 V electrical cabinets	A review of the “well-sealed” panels that house circuits below 440 V is in progress. The “well-sealed” panels represent a small percentage of the total Bin 15 count and will be removed from the Bin 15 frequency allocation. Additionally, the FPRA will be revised to include fire propagation from sealed panels > 440 V panels, consistent with the guidance in FAQ 14-0009.	Same approach as that used for the final integrated analysis provided in support of the LAR.

RAI No. / Description	Disposition with respect to the final integrated analysis and the aggregated results provided in support of the LAR (to be incorporated into the final PRA RAI 03 response which includes the final Fire PRA quantification results)	Disposition with respect to the self-approval model for post-transition changes
PRA RAI 8 regarding use of the transient frequency adjustment factors	Transient Frequency Adjustment Factor of 0.1 is being removed from the analysis and replaced with a frequency adjustment that is consistent with FAQ 12-0064.	Same approach as that used for the final integrated analysis provided in support of the LAR.
PRA RAI 9 regarding fire propagation in the MCR	Fire propagation in the main Control Room (MCR) is being addressed consistent with the guidance of NUREG/CR-6850 Chapter 11 and Appendix S.	Same approach as that used for the final integrated analysis provided in support of the LAR.
PRA RAI 11 regarding crediting MCR abandonment	The FPRA method for control room abandonment evaluation of the variant and compliant cases is addressed in the RAI response. No changes to this methodology are expected to be required, however, the values provided in the RAI response may change once revisions are incorporated and the final results are quantified.	Same approach as that used for the final integrated analysis provided in support of the LAR.
PRA RAI 12 regarding multiple versus single cables	The updated quantification will assume the heat release rates associated with multi-bundle configuration for all MCR panels.	Same approach as that used for the final integrated analysis provided in support of the LAR.
PRA RAI 14	See PRA RAI 15.	See PRA RAI 15.
PRA RAI 15 [corrected RAI number for this subject, original list incorrectly identified RAI as PRA RAI 14] regarding large reduction credit for modifications	The response to this RAI will calculate the total risk increase associated with the unresolved variances from deterministic requirements (VFDRs) (i.e., VFDRs that are not associated with a plant modification and discuss the impact of important modeling assumptions contributing to the risk significant scenarios for fire areas in the compliant plant model.	Not applicable to the development of the post transition self-approval model, since this RAI did not impact the integrated analysis and the aggregated results provided in support of the LAR.

RAI No. / Description	Disposition with respect to the final integrated analysis and the aggregated results provided in support of the LAR (to be incorporated into the final PRA RAI 03 response which includes the final Fire PRA quantification results)	Disposition with respect to the self-approval model for post-transition changes
FM RAI 1.k regarding evaluation of MCR abandonment times	The abandonment frequency will be updated as necessary to reflect the response provided for Fire Modeling (FM) RAI 1.k.	Same approach as that used for the final integrated analysis provided in support of the LAR.

Please refer to PRA RAI 17 for a discussion of focused-scope peer reviews performed to date and the process that ensures additional focused-scope peer reviews are performed against any model changes that constitute a PRA method upgrade.

All RAI responses requiring model changes will be captured in a tracking database on the Entergy SharePoint site to ensure that each issue is captured in the final response to PRA RAI 03. Changes that constitute a PRA upgrade, if any, will be identified and a focused-scope peer review will be initiated with associated Findings and Observations (F&Os) resolved in accordance with Entergy PRA procedures.

c) Final risk quantification will be provided after NRC concurrence is obtained with the approach described in the various RAI responses. This will ensure the integrated response reflects NRC accepted approaches.

No new modifications or operator actions are expected to be required to support the final FPRA quantification. Should any new modifications or operator actions be identified, such will be reflected in a markup of LAR Attachments S and G.

d) No unacceptable methods or weakness will be retained in the FPRA. Please refer to the table above for disposition of potentially unacceptable methods or weaknesses.

<u>PRA RAI 10</u> – MCR Abandonment Scenarios due to Loss of Habitability

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 Fire PRA 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. By letter to NEI dated July 12, 2006, the NRC established the ongoing FAQ process where official agency positions regarding acceptable methods can be documented until they can be included in revisions to RG 1.205 or NEI 04-02. 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 fire scenario analysis indicates that MCR abandonment due to loss of habitability (LOH) in the MCR was credited in the Fire PRA. However, beyond presenting the times to abandonment due to LOH for different fire scenarios, the analysis does not indicate how these scenarios were modeled. Table W-1a of the LAR appears to indicate that a single scenario (i.e., 129-F/A) was used to model MCR abandonment due to LOH based on an aggregation of the frequency of abandonment into a single frequency representing multiple fire sources in the MCR. There appears to be just five HFEs in the HRA designated as MCR abandonment actions which is in contrast to the large number of actions identified in the MCR abandonment procedure. Accordingly, it is not completely clear how MCR abandonment was modeled in the Fire PRA or how the range of potential fire-induced failures resulting from fires leading to MCR abandonment were addressed. In light of these observations, please provide the following information:

a) Please describe how MCR abandonment was modeled for LOH. Include an explanation of how the CCDPs and conditional large early release probabilities (CLERPs) were estimated. Include identification of the actions required to execute safe alternate shutdown and how they are modeled in the Fire PRA, including actions that must be performed before leaving the MCR.

b) Please explain how the CCDPs and CLERPs that were estimated for fires which lead to abandonment due to LOH address various possible fire-induced failures and provide the range of CCDPs for MCR abandonment due to LOH. Specifically include in this explanation, a discussion of how the following scenarios are addressed:

    i. Scenarios where fire fails only a few functions aside from forcing MCR abandonment and successful alternate shutdown is straightforward;

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

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

c) Please provide the frequency of MCR abandonment for LOH for the variant case and the compliant case.

d) Please provide the CCDP(s) and CLERP(s) for the variant case and the compliant case.

Response

a) Multiple scenarios were modeled for the various ignition sources within the MCR, however, only a single MCR abandonment fire scenario was quantified. This abandonment scenario calculated the CCDP and CLERP assuming all cables in the MCR were damaged and operator actions were credited to isolate the Reactor Coolant System (RCS) and start primary to secondary heat removal. The operator actions credited are specified in license amendment request (LAR) Attachment G for Fire Area G. These actions are:

1. Alignment and control of the new Common Feedwater (CFW) pump at its local control station, outside the MCR, using its associated HEP (developed using the NUREG-1921 methodology); see LAR Attachment S, Table S1, modification S1-1.
2. Tripping of the Reactor Coolant Pumps (RCPs) at the 6.9 kV switchgears.
3. De-energizing and closing the letdown isolation valve.
4. De-energizing the pressurizer Electromagnetic Relief Valve (ERV) at its power supply.
5. De-energizing the DC control power and tripping the primary makeup pumps at the respective 4 kV switchgear.

The HEPs for each of the above actions were revised to use the NUREG-1921 methodology, post LAR submittal. The results, to be provided in conjunction with PRA RAI 03, will use these NUREG-1921 based HEPs.

An operator action to trip the reactor prior to MCR abandonment is also credited.

b) The compliant case CCDP and CLERP are based on a review of VFDRs and elimination of the associated fire induced hardware failures to allow only random failures of the corresponding components to define the scenario risk. Operator actions credited in the compliant case, such as the tripping of an RCP, are assumed to be mitigated from within the MCR. The variant case CCDP and CLERP are based on failure of all cables in the MCR with a success path mitigated within the MCR and mitigated by the operator recovery actions listed in the response to a) above.

The use of a single MCR abandonment scenario to define the delta risk provides a bounding risk, and delta risk, estimate that is based on the assumption that all cables in the MCR are damaged. In this worst-case damage scenario the delta risk is maximized since the credit given to the VFDRs maximizes the offset of the risk of the variant case, which includes all VFDRs in the respective variant condition. Evaluation of single scenarios with less damage would result in a lower variant risk, but the same compliant risk, since credit is taken for an ideal primary control station for which the risk is not dependent on the MCR damage incurred. This approach is similar to the quantification of a scenario where the fire-induced failures cause great difficulty for shutdown by failing multiple functions and/or complex spurious operations that make successful shutdown unlikely (Item iii of the RAI). The evaluation of

i. scenarios where the fire fails only a few functions aside from forcing MCR abandonment and successful alternate shutdown is straightforward, and
ii. scenarios where the fire could cause some recoverable functional failures or spurious operations that complicate the shutdown, but successful shutdown is likely,

will reflect more limited damage, which will result in a smaller delta risk, since in the limit the delta is zero for a scenario that causes no damage. The primary advantage in applying a scenario-by-scenario MCR abandonment analysis for evaluation of delta risk is that a more accurate, but less conservative, delta risk results from each scenario evaluated. Therefore, the current methodology provides a bounding delta risk value. In addition, the large negative delta risk value for the ANO-1 FPRA due to the incorporation of significant modifications that are not required for a compliant plant will offset the most conservative estimate of the delta risk that can be obtained, one in which the compliant case risk is assumed to be zero (i.e., the MCR abandonment variant case CDF is expected to be less than the absolute value of the plant delta risk, which is a negative value). The final risk quantification values will be provided in the response to PRA RAI 03.

c) The frequency of abandonment is being updated in order to address fire modeling RAIs associated with the calculation of MCR abandonment times. The final value will be provided in the response to PRA RAI 03. The frequency of abandonment for the variant case and the compliant case are the same. The current frequency value submitted with the LAR is 4.35E-05/rx-yr, however, a new value will be provided in an updated Table W-1a, for scenario 129-F/A. Loss of control significant enough to require MCR abandonment is only expected to occur as a result of loss of habitability. In all non-loss of habitability cases, command and control for post fire shutdown is expected to remain in the MCR.

d) The CCDP and CLERP for the above referenced scenarios will also be provided with the response to PRA RAI 03 in Tables W-1a (CCDP) and W-1b (CLERP). The compliant case CCDP/CLERP for the MCR abandonment scenario associated with the LAR submittal quantification are 7.98E-02/1.13E-02. The variant case CCDP/CLERP for the MCR abandonment scenario associated with the LAR submittal quantification are 1.08E-01/1.44E-02.

The above RAI response is consistent with the response to ANO-2 PRA RAI 01.e.01 transmitted to the NRC in letter dated August 7, 2014 (2CAN081401, ML14219A635).

<u>PRA RAI 11</u> – Credit for MCR Abandonment for Loss of Control Scenarios

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 fire PRA 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. By letter to NEI dated July 12, 2006, the NRC established the ongoing FAQ process where official agency positions regarding acceptable methods can be documented until they can be included in revisions to RG 1.205 or NEI 04-02. 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 LAR does not indicate whether MCR abandonment (i.e., "evacuation" in procedure 1203.002) due to loss of control (LOC) (i.e. use of "Alternative Shutdown") is modeled or credited in the Fire PRA. Please clarify whether MCR abandonment due to LOC is credited in the Fire PRA analysis. If it was credited, then provide the following information:

a) Please explain how the variant plant PRA models scenarios which threaten "immediate damage to major portions of vital controls" or "immediate damage to a significant number of cables."

b) Please explain how the compliant plant PRA models scenarios which threaten "immediate damage to major portions of vital controls" or "immediate damage to a significant number of cables."

c) Please describe how the human error probabilities (HEPs) are developed for the scenarios in a) and b) above and confirm that the timing associated with the cues to implement alternate shutdown and the times available to perform alternate shutdown have been evaluated.

d) Please explain how the CCDPs/CLERPs are estimated for scenarios in a) and b) above.

e) Please provide your range of CCDPs/CLERPs for the scenarios in a) and b) above.

Response

a) In the FPRA model, the MCR scenarios for the variant plant consist of two types. Only one abandonment scenario is based on the frequency of abandonment, which conservatively assumes all cables routed through the MCR are damaged. Non-abandonment scenarios are evaluated for each panel, or group of panels, which communicate with each other. Loss of control significant enough to require MCR abandonment is only expected to occur as a result of loss of habitability. In all non-loss of habitability cases, command and control for post fire shutdown is expected to remain in the MCR. This is achievable given that the new CFW pump can be operated from the MCR or the pump's local control station, and will significantly reduce the required number of Operator actions required to stabilize the plant following a MCR or cable spreading room fire. In all other cases, where the MCR is habitable and the fire impact is limited to a single panel or a limited group of panels, the abandonment procedure is not expected to be initiated. The new CFW pump and associated Operator actions, along with other actions required to mitigate potential multi-spurious operations (MSOs), will be credited while the Operators remain in the MCR for the non-abandonment scenarios.

b) The compliant plant FPRA models the same scenarios specified in Item a) above, but the compliant variation assumes that the failures identified in the VFDRs are not impacted by fire (i.e., the compliant case simulates a scenario that assumes the cables for the VFDR components have been rerouted outside Fire Area G, or are otherwise protected, such that the cables are not impacted by a fire in Fire Area G). Additionally, the compliant configuration credits Operator actions, such as tripping the Reactor Coolant Pumps, in the MCR. Conversely, the variant case would credit a local recovery action.

c) The HEPs for actions required by scenarios discussed in Items a) and b) above are based on detailed HEPs developed considering the actual action time window and time frames for performance of the actions. For the compliant case, the same actions are modeled as the actions in the MCR / primary control station to reflect a lower compliant plant HEP. For the variant case, actions credited outside the MCR will have a higher failure probability due to the difference in conditions, timing, etc., associated with abandoning the MCR and performing the Operator recovery actions.

d) The CCDPs/CLERPs for the scenarios in Items a) and b) above are determined based on failure of the associated cables/components (all MCR cables and components for the abandonment scenario and cables and components located within the impacted panels for the non-abandonment scenarios) and credit for the associated Operator actions (with the HEPs calculated as discussed in Item c) above) for the variant and compliant case.

e) The range of CCDPs and CLERPs for the variant and compliant cases for the MCR fire scenarios, based on the model used in the LAR submittal, is provided below (note that the following values may change when the FPRA is re-quantified in response to PRA RAI 03):

MAIN CONTROL ROOM ABANDONMENT CASE (as described above, only one bounding abandonment scenario is quantified, in Fire Zone 129-F)

VARIANT CASE

CCDP	1.08E-01
CLERP	1.44E-02

COMPLIANT CASE

CCDP	7.98E-02
CLERP	1.13E-02

NON-ABANDONMENT CASE FOR MAIN CONTROL ROOM FIRES (Fire Zone 129-F)

	Minimum (Best Case Scenario)	Maximum (Worst Case Scenario)
VARIANT CASE		
CCDP	4.36E-07	1.84E-02
CLERP	2.89E-09	2.44E-03
COMPLIANT CASE*		
CCDP	1.64E-06	4.13E-02
CLERP	2.03E-07	7.22E-03

* values specified are calculated values; zero values are used for some scenarios to provide bounding delta risk

PRA RAI 13 – Calculation of VFDR ΔCDF and ΔLERF

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 FPP 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 and 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.

Attachment W of the LAR, Section W.2, "Risk Change due to NFPA 805 Transition," provides a description of how the change-in-risk associated with VFDRs is determined but not enough detail to make the approach completely understood. Please provide the following information:

a) A detailed definition of both the post-transition and compliant plant models used to calculate the reported change-in-risk due to the NFPA 805 transition. Include a description of the model adjustments made to remove VFDRs from the compliant plant model, such as adding events or logic, or use of surrogate events. Also, include an explanation of how VFDR and non-VFDR modifications are addressed for both the post-transition and compliant plant models. Include an explanation of whether the approach is consistent with guidance in FAQ 08-0054, "Demonstrating Compliance with Chapter 4 of NFPA 805."

b) A separate description of how the change-in-risk was determined for 1) MCR abandonment due to loss of habitability and 2) MCR abandonment due to loss of control. Include a discussion of how the CCDPs/CLERPs were determined for both the variant plant and the compliant plant models for each of these types of scenarios. 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 methods apply 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. Based on the modeling of the compliant and the variant plant, discuss whether the change-in-risk estimate is realistic, potentially conservative or potentially non-conservative.

c) An explanation of any major changes made to the Fire PRA model or data for the purpose of evaluating VFDRs.

d) A description of the type of VFDRs identified, and discussion whether and how the VFDRs identified, but not modeled in the Fire PRA, impact the risk estimates. Note 1 to Table W-2 of the LAR provides qualitative rationale for excluding VFDRs from the change-in-risk calculations associated with HVAC systems and equipment required to transition to cold shutdown. Identify and describe any other types of VFDRs excluded from the change-in-risk calculations.

Response

a) A review of the VFDRs generates a compliant plant model where the VFDR is no longer variant and complies with deterministic requirements. The VFDRs are defined by the deterministic analysis which identifies the performance deficiencies that do not meet deterministic requirements. The compliant case was based on the current plant design and operation with VFDRs removed from the model and without credit for the new CFW pump modification since it is implemented to reduce risk and represents a plant configuration which reduces risk beyond the requirements of a compliant plant. The VFDRs are removed from the compliant case model by setting the failure probability of VFDR affected components to the random failure probability of the component. The compliant configuration is generally defined by the elimination of VFDRs and not the incorporation of the risk reduction modifications.

The post-transition case was based on the anticipated plant design and operation including all planned modifications and all retained VFDRs. In the post-transition plant model, the failure probability of components affected by retained VFDRs is set to "failed by the fire" (i.e., probability 1.0). VFDRs that are eliminated by modifications are removed from the post-transition model by modeling the new configuration, which eliminates the failed-by-fire failure mode.

The methodology used to define the compliant case and calculate the delta risk is consistent with the guidance provided in FAQ 08-0054.

b) Please refer to the response to PRA RAIs 10 and 11 for a discussion of the compliant and variant plant configurations for MCR abandonment and non-abandonment scenarios. Loss of control significant enough to require MCR abandonment is only expected to occur as a result of loss of habitability. In all non-loss of habitability cases, command and control for post-fire shutdown is expected to remain in the MCR. This is achievable given that the new CFW pump can be controlled from the MCR and will significantly reduce the required number of operator actions required to stabilize the plant following a MCR or cable spreading room fire. For the MCR abandonment compliant configuration, the CCDP and CLERP reflect a configuration where the equipment relating to the VFDRs are set to their nominal failure probabilities and the operator actions are assumed to be taken from the MCR. The applied method ensures conservatism in the delta risk calculation by applying consistent assumptions for both the variant and compliant cases with the primary difference associated with the elimination of VFDRs in the compliant plant model and inclusion and operating the new CFW pump only in the post-transition model, in conjunction with the recovery actions listed in PRA RAI 10, Part a). Please refer to Item a) above for further discussion.

c) The only changes made relevant to VFDR evaluations are those described in Item a) above. These changes were limited to the use of altered events to eliminate VFDR configurations from the compliant plant model.

d) The following are some examples of the types of VFDRs that are identified for the Fire Risk Evaluation (FRE) performed in determining the delta risk:

Pressure Control	Loss of control and spurious opening of ERV block valve CV-1000 can result in RCS depressurization.
Inventory	Loss of control to letdown block valve CV-1221 (IN 92-18) preventing isolation of the RCS.
Decay Heat Removal	Loss of control and spurious closure of Steam Generator (SG)-B to Emergency Feedwater (EFW) pump P-7A steam block valve CV-2617 (IN 92-18).
Vital Auxiliaries – Electrical	Loss of control and the inability to trip breaker A-409 preventing transfer to Emergency Diesel Generator (EDG) #2 (K-4B).

Vital Auxiliaries – Service Water	Loss of control for valve CV-3643 (IN 92-18) resulting in a diversion of Service Water (SW) to Auxiliary Cooling Water (ACW).

The VFDRs that are not modeled in the FPRA are associated with functional requirements identified in the deterministic analysis. The functional requirements are not related to specific core damage sequences in the FPRA model and, therefore, do not impact the core damage frequency. The most notable VFDRs excluded from the change in risk calculations were those failures associated with Pressurizer control and failures associated with overcooling of the RCS. Additionally, an evaluation of MSOs was performed and revisions to the FPRA model were implemented to ensure that the impact of MSOs with respect to potential core damage sequences was included in the FPRA model in order to allow quantification of delta risk associated with VFDRs related to MSOs.

PRA RAI 15 – Large Reduction Credit for Modifications

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 FPP 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 and 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.

Appreciable risk reduction credit is presented in Attachment W of the LAR for non-VFDR risk reduction modifications, judging by the large negative total change-in-risk. Section 3.2.5 of RG 1.205 states that risk decreases may be combined with risk increases for the purposes of evaluating combined changes in accordance with regulatory positions presented in Sections 1.1 and 1.2 of RG 1.174, Revision 2, but that the total increase and total decrease in the ΔCDF and ΔLERF should be provided. Table W-1a of the LAR reports the net change-in-risk values for each fire area after the risk increase associated with unresolved VFDRs is offset with the risk decrease associated with modifications, but does not present the total increase associated with unresolved VFDRs and the total decrease associated with modifications.

The NRC staff notes that conservative calculation of the compliant plant CDF and LERF can lead to a non-conservative calculation of the ΔCDF and ΔLERF. The NRC staff observes that the dominant scenario listed in Table W-1a of the LAR (i.e., 100-N/A) involves a scenario in which all targets in the fire zone are damaged which seems to imply the scenario was conservatively modeled. Dispositions to Fire PRA F&Os indicate that modeling conservatisms exist such as limited credit for suppression and detection systems and simplified modeling for high risk areas. The NRC staff acknowledges that installation of a new auxiliary feedwater (AFW) pump, redundant DC control power to switchgear, and other modifications represent significant risk reduction. However, given the significance of the risk reduction credited for modifications, and the fact that a conservative calculation of the compliant plant CDF and LERF can lead to a non-conservative calculation of the ΔCDF and ΔLERF, please provide the following information:

a) Please provide (or point out if already provided) the total risk increase associated with unresolved VFDRs and the total decrease associated with non-VFDR modifications.

b) Please summarize the risk significant scenarios for fire areas in the compliant plant model which are most significantly impacted by risk reduction from modifications, and discuss the contribution of fire-induced failures for those scenarios.

c) Please discuss the impact of any important modeling assumptions contributing to the risk significant scenarios for fire areas in the compliant plant model. Specifically address conservative modeling assumptions made in the compliant plant model that may artificially reduce the calculated change-risk-risk (or result in overestimating the risk offset).

d) If conservative modeling of the compliant plant contributes to the under estimation of the total change in risk, then demonstrate that the total risk increase associated with unresolved VFDRs is offset by the total risk decrease associated with risk reduction modifications even when the conservative modeling is removed. Alternatively, demonstrate that the conservative modeling does not significantly impact the change in risk or replace the conservative modeling with realistic modeling that does not underestimate the total change-in-risk in the integrated analysis provided in response to PRA RAI 3.

Response

a) The information requested in this RAI can be extracted from the FPRA results that were submitted with the LAR. However, due to the concurrent changes in the modeling to address the PRA RAIs, this information will vary once the model changes have been implemented. Therefore, to provide an accurate response for the requested risk increase/decrease, the requested results will be provided in conjunction with PRA RAI 03.

To calculate the total risk increase associated with the unresolved VFDRs (i.e., VFDRs that are not associated with a plant modification), the results will be generated by incorporating all of the risk reduction mods into the compliant configuration. It should be noted that the redundant DC modification is currently in the compliant configuration and has relatively little quantification impact on the complaint case. The CFW modification is the most significant modification in the ANO-1 model that contributes to the negative offset reported in the ΔCDF and ΔLERF values. The CFW pump is being uniquely designed for mitigating severe fire accidents and can be utilized if EFW is deemed unavailable due to failures in fire zones, especially those that contain a significant number of safe shutdown cables (e.g., the cable spreading room). Therefore, to assess the risk decrease associated with the addition of the CFW system, and the risk increase of unresolved VFDRs, a sensitivity analysis will be performed and will be provided in the response to PRA RAI 03. The sensitivity analysis will include the CFW pump in the compliant configuration, which effectively mitigates the decay heat removal VFDRs. By comparing the sensitivity results to the variant case results, the delta risk will represent the risk of the unresolved VFDRs that have been offset by the greater risk decrease of the CFW modification. The risk offset from considering the CFW pump in the compliant case will be calculated by subtracting the ‘risk increase’ of the unresolved VFDRs from the total absolute value of the delta risk presented in the revised Attachment W (associated with the PRA RAI 03 response). In summary, this calculation would separately provide the total risk increase of the unresolved VFDRs and the risk offset provided by the CFW pump to achieve the total delta CDF/LERF, to be provided in the updated Attachment W. The results will be provided using the approach applied by the NRC in the ANO-2 NFPA 805 Safety Evaluation (SE) to ensure that the data is in the form required to support the ANO-1 SE.

b) As discussed in the response to Item a) above, due to the changes in the model, the requested summary will be provided in conjunction with the response to item a) with the PRA RAI 03 response. This will ensure the most updated response with respect to the result reported in the ANO-1 SE.

c) For each compliant case scenario, the VFDRs were reviewed against the FPRA to determine if the function relating to the VFDR was modeled. For those VFDRs which were modeled within the FPRA, the affected components were set to their nominal value (instead of failed) for quantification of the compliant case. This approach was utilized consistently for all fire areas analyzed. Additionally, the fire modeling will utilize the acceptable approaches as defined in NUREG/CR-6850 and the approved FAQs. These approaches minimize potential conservatisms in the ANO-1 FPRA model results. The methodology used in the ANO-1 FPRA analysis is similar to that performed for the ANO-2 FPRA. The two main conservatisms identified in the ANO-2 FPRA model, and addressed in the ANO-2 SE, also apply to ANO-1. These two main conservatisms are the assumed failure of components with unknown routing, and a single conservative MCR abandonment estimate.

   Although the FPRA modeling associated with components for which cable routing was not performed is conservative, this same assumption was made in both the compliant and variant cases, thus providing some offset to the impact of this conservative assumption. In relation to the MCR abandonment case, the single abandonment estimate was also made for both the compliant and variant cases. These two assumptions will be addressed as part of the analysis to be performed in response to Item d) below.

d) The conservatisms discussed in Item c) above will be reviewed to demonstrate that the total risk increase associated with the unresolved VFDRs is offset by the total risk decrease associated with risk reduction modifications despite conservative assumptions in the compliant case configurations. It should be noted that for the large severe fires, the ANO-1 plant modification and recovery actions are sufficient to mitigate the most severe and consequential cases. However, it is understood that a concern exists regarding the conservatisms in the compliant case results affecting the overall delta risk calculations. This can only be true for negative ΔCDF/LERF calculations due to an unproportioned offset from the new CFW system. The results from Item a) above will remove all offset from the added CFW system in the delta risk calculation and will provide a bounding ΔCDF/ΔLERF analysis for the remaining unresolved VFDRs. It is noted that the CFW system also provides a risk offset for scenarios with no decay heat removal performance deficiencies. The results from Item a) above will be evaluated to ensure that the delta CDF/LERF do not meet Regulatory Guide 1.174 solely as the result of conservative assumptions in the compliant case that could result in an overestimation of the risk offset (decrease) in the FPRA model. The FPRA model is being concurrently updated with the RAI responses and a follow-up response will be required within the results associated with the PRA RAI 03 response. The results of this evaluation will be characterized in a manner similar to that provided by the NRC in the ANO-2 NFPA 805 SE to facilitate incorporation into the ANO-1 SE.

PRA RAI 17 – Focused Scope Reviews and Model Changes after the Full Peer Review

NFPA 805 Section 2.4.3.3 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 fire PRA 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. RG 1.200 (currently Revision 2) 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.

Attachment U of the LAR states that a "Regulatory Guide (RG) 1.200, Revision 1 Peer Review against the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard requirements" was performed on the ANO-1 Internal Events PRA in August 2009. Revision 2 of RG 1.200 was issued in March 2009, and is the version of the regulatory guide that should be used to support risk-informed applications submitted after March 2010, including the ANO-1 NFPA-805 LAR. The NRC staff also notes that a number of revisions and updates were made in response to the reported peer review F&Os.

a) Clarify how the review of the ANO-1 PRA complies with Revision 2 of RG 1.200 and provide any additional F&Os or related findings summarizing the assessment of the ANO-1 Internal Events PRA against Revision 2.

b) If any changes made to the Internal Event PRA or Fire PRA since the last full-scope peer review are consistent with the definition of a PRA upgrade in ASME/ANS-RA-Sa-2009, confirm whether a focused-scope peer review was performed for these changes. Provide any findings from that focused-scope peer review and the resolution of these findings, unless they were already provided in the LAR.

c) Confirm that all focused-scope peer reviews were performed consistent with RG 1.200 endorsed methods.

Response

a) The statement in the LAR, "The ANO-1 PRA has undergone a Regulatory Guide (RG) 1.200, Revision 1, Peer Review against the American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) PRA Standard requirements," incorrectly refers to Revision 1 of RG 1.200. The Peer Review actually used Revision 2 of RG 1.200. The specific guidance used in the Peer Review was: 1) NEI 05-04, "Process for Performing Follow-on PRA Peer Reviews Using the ASME PRA Standard," Nuclear Energy Institute, Rev. 2, November 2008; 2) "Standard for Level1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME/ANS RA-Sa-2009, February, 2009; and 3) NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Rev. 2, March 2009.

b) Focused-scope peer reviews were performed for FPRA ASME/ANS Standard Elements FSS (see LAR Attachment V for a discussion of the two FSS related focused-scope peer reviews) and HRA (as described in PRA RAI 01.g). No additional changes to the methods are anticipated that would be considered a PRA upgrade. Should any such changes be identified as the final FPRA quantification is completed, the Entergy procedures governing the need for focused scope peer reviews would trigger an additional FPRA focused-scope peer review. Findings from the focused-scope peer reviews performed to date were included in the LAR for the FSS element focused-scope peer reviews. The HRA focused-scope peer review was provided to the NRC audit team during the April 2015 NFPA 805 audit.

c) All three focused scope peer reviews performed to date were performed consistent with RG 1.200 endorsed methods.

<u>PRA RAI 18</u> – Minimum Joint HEP Values

Confirm that each joint HEP value used in the Fire PRA below 1.0E-05 includes its own justification that demonstrates the inapplicability of the NUREG-1792 lower value guideline. Provide an estimate of the number of these joint HEPs below 1.0E-05 and at least two different examples of the justification.

Response

The FPRA will be re-quantified using a joint HEP value that is greater than or equal to 1.0E-05. These results will be provided in response to PRA RAI 03. If during the review of the results it is identified that this conservative assumption results in a significant departure from realism, the HEPs having a high importance would be revisited and, if a value less than 1E-05 is calculated, specific justification will be provided. The primary basis for justification of combination events of less than 1.0E-05 will be associated with joint HEPs comprised of independent actions that are significantly separated in time and for which necessary cues are available. The results of this review and associated joint HEPs credited will be available for NRC review post PRA RAI 03 in the FPRA Human Reliability Analysis notebook (PRA-A1-05-007).

REFERENCES

1. Entergy letter dated January 29, 2014, *License Amendment Request to Adopt NFPA-805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)* (1CAN011401) (ML14029A438)

2. NRC letter dated May 5, 2015, *Arkansas Nuclear One, Unit 1 – Request for Additional Information Regarding License Amendment Request to Adopt National Fire Protection Association Standard 805* (TAC No. MF3419) (1CNA051501) (ML15091A431)

3. Entergy letter dated May 19, 2015, *Response to Request for Additional Information – Adoption of National Fire Protection Association Standard NFPA-805* (1CAN051501) (ML15139A196)

4. Entergy letter dated June 16, 2015, *60-Day Response to Request for Additional Information – Adoption of National Fire Protection Association Standard NFPA-805* (1CAN061501) (ML15167A503)

5. Entergy letter dated July 21, 2015, *90-Day Response to Request for Additional Information – Adoption of National Fire Protection Association Standard NFPA-805* (1CAN071501) (ML15203A205)