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Omaha Public Power District
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Omaha, NE 68102-2247

LIC-13-0142
October 11, 2013

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

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40
NRC Docket No. 50-285

References: See Reference List on Page 3

SUBJECT: Probabilistic Risk Assessment (PRA) RAI Response - NFPA-805 Transition (ME7244)

The Omaha Public Power District's (OPPD's) response to the Nuclear Regulatory Commission's (NRC's) request for additional information (RAI) regarding the probabilistic risk assessment (PRA) is provided in the attachment to this letter. As noted in the Reference 11 email, this RAI is from the second set of the third round of RAIs regarding the license amendment request (LAR) to adopt National Fire Protection Association (NFPA) 805 at the Fort Calhoun Station (FCS).

In the Reference 1 LAR, OPPD requested an amendment to Renewed Facility Operating License No. DPR-40 for FCS, Unit No. 1, to adopt NFPA 805, *Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition)*. The NRC staff reviewed OPPD's application and determined that additional information was required in order to complete their review and subsequently transmitted RAIs via References 2, 6 and 9. OPPD provided responses to these RAIs in References 3, 4, 5, 7, 8, and 10. The NRC indicated that the staff had reviewed the information provided by the licensee [in References 3, 4, 5, 7, 8, and 10] and determined that additional information specified in the Reference 11 email is needed for the staff to complete its review.

As requested by Reference 11, the attachment to this submittal contains responses to PRA RAIs 01.i.02, 01.j.02, 01.j.03, 23.01, and 27. The attachment also contains responses to PRA RAIs 07.02 and 25. A response to PRA RAIs 19.01, 24, and 26 will be provided by November 6, 2013.

In Reference 12, OPPD responded to Safe Shutdown Analysis (SSA) RAI 17.01 by providing a circuit design sketch and description of the modification. As discussed in Reference 13, the NRC staff reviewed that material and determined that additional information is required in order to complete its review. OPPD agreed to provide a response to Reference 13 by October 25, 2013. There are no new regulatory commitments being made in this letter because of the enclosed NFPA 805 RAI responses. Please note, as indicated in References 3, 4, 7, 8, 10, and 12, OPPD plans to supplement the NFPA 805 transition LAR, which will reflect the applicable information delineated in the enclosed RAI responses. AR 48249 is tracking the LAR supplement.

In accordance with 10 CFR 50.91, a copy of this letter, without the attachment, is being provided to the designated State of Nebraska official.

If you should have any questions regarding this submittal or require additional information, please contact Mr. Bill R. Hansher, Supervisor-Nuclear Licensing, at 402-533-6894.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 11, 2013.

Louis P. Cortopassi
Site Vice President and CNO

LPC/BJV/mle

Attachment: Response to PRA RAIs 01.i.02, 01.j.02, 01.j.03, 07.02, 23.01, 25, and 27

c: S. A. Reynolds, Acting NRC Regional Administrator, Region IV
J. W. Sebrosky, NRC Senior Project Manager
L. E. Wilkins, NRC Project Manager
J. C. Kirkland, NRC Senior Resident Inspector
Manager Radiation Control Program, Nebraska Health & Human Services, R & L Public Health Assurance, State of Nebraska (w/out attachment)

Reference List

1. Letter from OPPD (J. A. Reinhart) to NRC (Document Control Desk), *License Amendment Request 10-07, Proposed Changes to Adopt NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition) at Fort Calhoun Station*, dated September 28, 2011 (LIC-11-0099) (ML112760660)
2. Letter from the NRC (L. E. Wilkins) to OPPD (David J. Bannister), *Fort Calhoun Station, Unit No.1 - Request for Additional Information Re: License Amendment Request to Adopt National Fire Protection Agency Standard NFPA 805 (TAC No. ME7244)*, dated April 26, 2012 (NRC-12-0041) (ML121040048)
3. Letter from OPPD (D. J. Bannister) to NRC (Document Control Desk), *Responses to Requests for Additional Information Re: License Amendment Request 10-07 to Adopt NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants," 2001 Edition, at Fort Calhoun Station*, dated July 24, 2012 (LIC-12-0083) (ML12208A131)
4. Letter from OPPD (D. J. Bannister) to NRC (Document Control Desk), *Responses to Requests for Additional Information Re: License Amendment Request 10-07 to Adopt NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants," 2001 Edition, at Fort Calhoun Station*, dated August 24, 2012 (LIC-12-0120) (ML12240A151)
5. Letter from OPPD (L. P. Cortopassi) to NRC (Document Control Desk), *Responses to Requests for Additional Information Re: License Amendment Request 10-07 to Adopt NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants," 2001 Edition, at Fort Calhoun Station*, September 27, 2012 (LIC-12-0135) (ML12276A046)
6. Email from NRC (L. E. Wilkins) to OPPD (D. L. Lippy), *DRAFT: Fort Calhoun NFPA 805, Second Round (ME7244)*, dated February 22, 2013 (NRC-13-0014)
7. Letter from OPPD (M. J. Prospero) to NRC (Document Control Desk), *Responses to Second Request for Additional Information Re: License Amendment Request to Adopt NFPA 805 at Fort Calhoun Station (TAC No. ME7244)*, dated April 23, 2013 (LIC-13-0033)
8. Letter from OPPD (L. P. Cortopassi) to NRC (Document Control Desk), *Remaining Responses to Second Request for Additional Information Re: License Amendment Request to Adopt NFPA 805 at Fort Calhoun Station (TAC No. ME7244)*, dated May 21, 2013 (LIC-13-0060)
9. Email from NRC (J. M. Sebrosky) to OPPD (D. L. Lippy), *Fort Calhoun NFPA 805, Third Round of RAIs (ME7244)*, dated June 27, 2013 (NRC-13-0081)
10. Letter from OPPD (L. P. Cortopassi) to NRC (Document Control Desk), *Responses to Third Request for Additional Information Regarding License Amendment Request to Adopt NFPA 805 at Fort Calhoun Station (TAC No. ME7244)*, dated July 29, 2013 (LIC-13-0096)
11. Email from NRC (L. E. Wilkins) to OPPD (D. L. Lippy), *Fort Calhoun NFPA 805, Third Round, Second Part, of RAIs (ME7244)*, dated August 14, 2013 (NRC-13-0102)
12. Letter from OPPD (L. P. Cortopassi) to NRC (Document Control Desk), *Safe Shutdown Analysis (SSA) RAI Response - NFPA-805 Transition (ME7244)*, dated September 12, 2013 (LIC-13-0129)
13. Email from NRC (L. E. Wilkins) to OPPD (M. L. Edwards), *Fort Calhoun Station Fourth Round RAIs for NFPA 805 LAR (ME7244)*, dated September 27, 2013

Omaha Public Power District (OPPD)
Response to PRA RAIs 01.i.02, 01.j.02, 01.j.03, 07.02, 23.01, 25, and 27
License Amendment Request to Adopt National Fire Protection Association Standard 805
Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants
at Fort Calhoun Station, Unit 1 (TAC No. ME7244)

By letter dated September 28, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112760660), as supplemented by letters dated December 19 and 22, 2011, and March 20, 2012 (ADAMS Accession Nos. ML113540334, ML11363A077, and ML12083A147, respectively), Omaha Public Power District, (the Licensee), submitted a license amendment request (LAR) to transition their fire protection licensing basis at the Fort Calhoun Station, Unit 1, from Title 10 of the Code of Federal Regulations (CFR), Section 50.48(b), to 10CFR50.48(c), National Fire Protection Association Standard NFPA 805 (NFPA 805). A review team, consisting of U.S. Nuclear Regulatory Commission (NRC) staff and contractors from Pacific Northwest National Laboratory (PNNL) and the Center for Nuclear Waste Regulatory Analyses (CNWRA) participated in a regulatory audit of Fort Calhoun in Blair, NE from March 5 -9, 2012. By letter dated April 26, 2012, (ADAMS Accession No. ML12198A406) the NRC issued requests for additional information (RAIs). By letters dated July 24, 2012 (ADAMS Accession No. ML12208A131), August 24,2012 (ADAMS Accession No. ML12240A151), and September 27,2012 (ADAMS Accession No. ML12276A046) the licensee provided responses to the RAIs. The NRC staff reviewed the information provided by the licensee in response to the first set of RAIs and determined that additional information was needed for the staff to complete its evaluation. Consequently, the staff issued a second round of RAIs on February 22, 2013, (ADAMS Accession No. ML13053A226) and a third round of RAIs on June 27, 2013 (ADAMS Accession No. ML13178A035). The licensee responded to these RAIs in letters dated April 23, 2013 (ADAMS Accession No. ML13116A015), May 21, 2013 (ADAMS Accession No. ML13144A814), and July 29, 2013 (ADAMS Accession No. ML13211A055).

The U.S. NRC staff has reviewed the information provided in your application and determined that additional information is required in order to complete its review. These RAIs can be found below. The NRC considers these RAIs to be the second set of the third round of RAIs. Based on discussions with you on August 13, 2013, it was agreed that a response to the RAIs found below will be provided in accordance with the following schedule:

- **Safe Shutdown Analysis (SSA) RAI response to be provided by September 13, 2013**
- **PRA RAIs 01.i.02, 01.j.02, 01.j.03, 23.01, and 27 responses to be provided by October 7, 2013**
- **PRA RAIs 07.02, 19.01, 24, 25, and 26 responses to be provided by November 6, 2013.**

In addition, as discussed with you during the August 13, 2013, phone call the staff has determined that you no longer need to provide a response to SSA RAI 07.01 that was issued to you on June 27, 2013 (ADAMS Accession No. ML13178A035). The staff determined that SSA RAI 07.01 response is not needed in order for the staff to complete its safety evaluation. The staff also discussed with you during the August 13, 2013, phone call that the response to PRA RAI 24 should include two additional sensitivity studies as a result of issues that were raised during a July 22 through July 24, inspection at your site.

Should the NRC determine that the RAIs found below are no longer necessary prior to the dates found above, the request will be withdrawn. If circumstances result in the need to revise the requested response date, please contact me or Joe Sebrosky.

Appendix A: PRA RAI 01.i.02

A.1 PRA RAI 01.i.02

By letter dated April 23, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13116A015) the licensee responded to RAI 01.i.01.b and stated that the change in risk reported from the sensitivity study on increased ignition frequencies resulted in lower (not the expected higher) change in risk values because new credit for human actions in alternative shutdown process (in abnormal operating procedure (AOP) 06) was simultaneously added into the PRA. No acceptable approach for the human reliability analysis (HRA) for abandonment due to loss of control has been developed and thus no technical basis for this new credit exists, (i.e., for the human error probability/conditional core damage probability (HEP/CCDP) for abandoning the main control room (MCR) due to loss of control. No description or justification for the HRA method was provided. Please provide the results of the requested sensitivity study on ignition frequencies by removing the credit of the new operator action.

Please clarify whether credit will be retained in the PRA for abandonment of the MCR on loss of control for fires in the cable spreading room (CSR), and, if so, describe and justify the HRA methods applied and their relation to current HRA methods. Currently, these actions are listed as being retained as defense-in-depth (DID) actions in FC41 per license amendment request (LAR) Attachment G (ADAMS Accession No. ML11376A072, non-publicly available) and not credited in the PRA. Update the information as necessary.

OPPD Response to PRA RAI 01.i.02

Response to PRA RAI 01.i documents a sensitivity study assessment of total plant CDF, total plant LERF, VFDR ΔCDF, and VFDR ΔLERF using the NUREG/CR-6850 generic fire frequencies for bins where the NUREG/CR-6850 Supplement 1 alpha values are less than or equal to one.

In subsequent response to PRA RAI 07.a, the FC41 cable spreading room was re-assigned to the Control/Aux/Reactor Building generic location, per the guidance of NUREG/CR-6850 Table 6-2. In the base FPRA supporting the PRA RAI 01.i sensitivity study, FC41 had been incorrectly assigned to the Plant-Wide generic location.

With FC41 re-assigned to the Control/Aux/Reactor Building generic location, there are no FC41 ignition sources whose NUREG/CR-6850 Supplement 1 alpha values are less than or equal to one. Therefore, FC41 is no longer applicable to the fire frequency sensitivity study (i.e., the FPRA results for FC41 are not sensitive to use of NUREG/CR-6850 frequencies in place of the NUREG/CR-6850 Supplement 1 frequencies for bins with alpha values less than or equal to one).

The impact of not crediting the alternate shutdown process to mitigate FC41 cable spreading room fires will be assessed as requested in response to PRA RAI 24 (currently scheduled to be submitted to the NRC on November 7, 2013). As stated in PRA RAI 01.i.02, currently "no acceptable approach for the human reliability analysis (HRA) for abandonment due to loss of control has been developed and thus no technical basis for this new credit exists." Accordingly, OPPD will remove from the NFPA 805 fire PRA credit to alternate shutdown for FC41 cable spreading room fires, and OPPD will not credit alternate shutdown due to loss of control caused by fires within the FC42 main control room, until a method acceptable to the NRC has been published. OPPD will retain procedural implementation of alternate shutdown for FC41 as defense-in-depth (DID) consistent with LIC-11-0099 Attachment G Step 5. **[AR 48249]**

Appendix B: PRA RAI 01.j.02

B.1 PRA RAI 01.j.02

By letter dated May 21, 2013 (ADAMS Accession No. ML13144A814), the licensee responded to RAI 01.j.01.a.iv and provided a HEP of 1.50E-02 and CCDP/conditional large early release probability (CLERP) of 1.41E-01 for abandoning the MCR due to loss of habitability with no explanation of how these values were developed.

a. **Please provide a summary of how the HEP of 1.50E-02 was developed. The discussion should include whether each of the performance shaping factors identified in Section 4.6 of NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," and if each of the large number of MCR abandonment recovery actions (RAs) identified in LAR Attachment G were addressed in the detailed HRA.**

b. **In light of the large number of primary control station (PCS) activities and RAs identified in LAR Attachment G, identify the actions that are the significant contributors to the HEP and discuss their timing and complexity. Also discuss the timing and complexity of the non-significant actions.**

OPPD Response to PRA RAI 01.j.02

Part 'a'

OPPD performed a detailed HRA of the alternate shutdown process, in accordance with NUREG-1921 Section 5.3 and Appendix B. A summary of OPPD's evaluation follows:

The analysis yielded a human error probability of 1.50E-02, which represents operator failure to prevent core damage using the alternate shutdown process. When equipment reliability is considered, the CCDP of control room abandonment is calculated as 1.41E-01. The CLERP is also taken to be 1.41E-01, since the alternate shutdown process does not include provision for containment isolation. Note that containment isolation may occur automatically following control room abandonment, but this plant response has not been explicitly evaluated. Because AOP-06 provides direction only to align one safe shutdown train, equipment reliability is the dominant contributor to the overall CCDP and CLERP.

First, the AOP-06 alternate shutdown procedure was reviewed for feasibility in accordance with the guidance in NUREG-1921 Section 4.3. This assessment considered time, manpower, cues, procedures and training, accessibility, equipment and tools, and operability of required components. The assessment concluded that the AOP-06 alternate shutdown process is feasible, as summarized in response to PRA RAI 01.j.01.a.i.

Next, a Human Failure Event (HFE) representing the alternate shutdown process was developed using the EPRI HRA Calculator, following the guidance of NUREG-1921 Appendix B. The HFE definition included twenty-eight (28) AOP-06 actions identified to be critical for preventing core damage. The actions primarily surrounded reactor trip, successful abandonment, transfer of control, establishing auxiliary feedwater, isolating the RCS and maintaining inventory. The identification of the critical actions were performed by the PRA analysts and confirmed via operator talk-through. The actions included in the HFE are functionally equivalent the LIC 11-0099 Table G-1 (listed under Fire Area 41/42) actions important for preventing core damage.

During HFE definition, fire impact on all of the NUREG-1921 Section 4.6 performance shaping factors was considered (i.e., cues and indications, timing, procedures and training, complexity, workload/pressure/stress, human-machine interface, environment, special equipment, special fitness needs, and crew communications / staffing / dynamics).

The cognitive portion of the HEP was quantified using the Cause-Based Decision Tree Method (CBDTM), and the execution portion of the HEP was quantified using the Technique for Human Error Rate Prediction (THERP).

Finally, operator interviews were performed to confirm the feasibility assessment, the selection of critical procedure steps, treatment of performance shaping factors, and the overall HRA results and conclusions.

Part 'b'

All twenty-eight (28) execution steps modeled by the control room abandonment HFE are significant contributors to the HEP. However, approximately half of the steps could involve errors of commission, and these therefore contribute more to the overall HEP calculation than those steps where error of commission is not relevant. For example, cycling FW-10 to maintain steam generator level could be performed incorrectly (error of commission), whereas manually tripping the reactor is not likely to be performed incorrectly (no error of commission due to obvious nature of reactor trip controls). In this analysis, the total failure probability of an individual procedure step involving error of commission is a factor of two (2) greater than a step not involving error of commission.

Regarding timing, "t=0" is the onset of conditions necessitating abandonment (i.e., loss of habitability or loss of a key safety function). Achieving a safe and stable end state via alternate shutdown is modeled to require completion within 60 minutes (T_{SW}). Control room evacuation is modeled to occur at five minutes ($T_{1/2}$), and control at the alternate shutdown panels is modeled to be established at 20 minutes (T_M). Operator interviews confirmed that control can be established within 15 minutes following control room evacuation. This leaves 40 minutes as the time available for recovery.

Regarding complexity, the overall HFE is quantified with execution complexity set to "complex." No aspect of the alternate shutdown process is modeled by this HRA to be "simple" in the HEP quantification.

OPPD recognizes that the NRC and industry are developing a more detailed HRA methodology for main control room abandonment via FAQ 13-0002. Following completion of this FAQ, OPPD will review the finalized method and implement as appropriate.

Appendix C: PRA RAI 01.j.03

C.1 PRA RAI 01.j.03

By letter dated May 21, 2013 (ADAMS Accession No. ML13144A814), the licensee responded to RAI 01.j.01.a.iv and indicated that the sensitivity analysis results reflect a change in the optical density criterion for abandoning the MCR due to loss of habitability from 0.3 m-1 to 3.0 m-1. Please clarify if the resultant abandonment time/probability bounds the NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," Section 11.5.2.11, heat flux criterion of 1 kW/m2 at 6' above the floor, corresponding to a smoke layer temperature of 95°C. If not, provide a sensitivity analysis of core damage frequency/large early release frequency (CDF/LERF), delta (Δ) CDF and Δ LERF from using the abandonment time/probability based on the heat flux criterion.

OPPD Response to PRA RAI 01.j.03

For each main control room fire simulation, abandonment time due to optical density and abandonment time due to heat flux were each determined. The most limiting abandonment time (between optical density and heat flux) was used in the calculation of non-suppression probability. In general, abandonment due to optical density occurred prior to abandonment due to heat flux. This is sensible given the relatively high soot yield used in the main control room fire simulations. Thus, the abandonment time calculations are bounding of NUREG/CR-6850, Section 11.5.2.11 criteria.

Appendix D: PRA RAI 07.02

D.1 PRA RAI 07.02

By letter dated May 21, 2013 (ADAMS Accession No. ML13144A814), the licensee responded to PRA RAI 07.01.d, and proposed an administrative limit to require a continuous fire watch when transient combustibles with the potential to damage targets are stored in the CSR, FC41. However, the process to determine when transient combustibles can cause damage, and thus require a continuous fire watch, was not explained. The response did imply that a permitting process already exists to establish when transient combustibles might damage targets. Please describe the permitting process and indicate how this process provides confidence that no targets would be damaged by a transient fire.

The response also indicated that this new proposed administrative limit is already modeled in the fire PRA (FPRA). Describe how the existing PRA captures this new proposed administrative limit. Specifically indicate if the severity factor applied in the analysis encompasses those heat release rates (HRRs) from transient combustibles which can cause damage, yet are less than the 5 pound combustible limit originally employed in FC41.

OPPD Response to PRA RAI 07.02

As described in response to PRA RAI 07.01.d, OPPD proposes revising SO-G-91 *Standing Order Control and Transportation of Combustible Materials*. This procedure revision would require a continuous fire watch when transient combustibles with the potential to damage targets are stored in FC41 Cable Spreading Room.

Section 5.1.2 of SO-G-91 will be revised to state: "No transient combustible material shall be left inside the Cable Spreading Room (Room 70) without evaluation and approval by the Fire Protection Engineer via Form FC-1244." Form FC-1244 documents the original request and Fire Protection Engineer evaluation. Approved FC-1244 forms are returned to the requestor and posted in the area where the combustible material is located. Upon completion of work activities, the FC-1244 form is returned to the Fire Protection Engineer for closeout and formally maintained as a QA record.

The Fire Protection Engineer is responsible for ensuring that combustible material permitted in FC41 is in accordance with the FCS license basis, which in this case means that no combustible material with the potential to damage targets shall be left in FC41 without a continuous fire watch. The determination of whether a proposed combustible package has the potential to damage targets will be made using established fire protection engineering data and methods, such as those documented in NUREG-1805, the *SFPE Handbook of Fire Protection Engineering*, the *NFPA Fire Protection Handbook*. The evaluation would consider factors such as the expected peak heat release rate, spatial geometry between the combustible and nearby targets, target damage temperatures and heat fluxes, and the various exposure mechanisms (i.e., flame emersion, flame radiant heat flux, plume, ceiling jet, and hot gas layer). In practice, due to the proximity of FC41 overhead cable trays to the floor, it is expected that this process would rarely allow anything more than a trivial / negligible transient combustible fuel package unattended in FC41. Procedural guidance to assist the Fire Protection Engineer in performing this evaluation will be developed, likely as an appendix to SO-G-91. **[AR 48249]**

The FPRA models the enhanced combustible control by crediting a continuous fire watch to provide prompt suppression for all FC41 transient fire scenarios. This is consistent with the enhanced combustible control, which requires a continuous fire watch for all combustible packages capable of damaging targets (i.e., capable of inducing an initiating event and/or failing mitigating equipment). The risk associated with failure to assign a continuous fire watch (i.e., violating the combustible control process) is quantified in response to PRA RAI 07.01.c.

PRA RAI 07.02 requests that OPPD "Specifically indicate if the severity factor applied in the analysis encompasses those heat release rates (HRRs) from transient combustibles which can cause damage, yet are less than the 5 pound combustible limit originally employed in FC41." In response, the severity factor for each FC41 transient fire does include contribution of all heat release rates physically capable of damaging targets, independent and regardless of the original five pound combustible limitation. This process recognizes that, under certain conditions, even combustibles less than five pounds could cause target damage.

In summary, OPPD will implement a formal process to ensure that no transient combustible material capable of damaging targets will be left inside the FC41 Cable Spreading Room without a continuous fire watch. This process is reflected in the FPRA by crediting a continuous fire watch for all FC41 transient fire scenarios. Response to PRA RAI 07.01.c evaluates the contribution of failing to comply with the combustible control process. Finally, FC41 transient fire severity factors and non-suppression probabilities consider the full range of heat release rates capable of damaging targets (regardless of the original five-pound limitation).

Appendix F: PRA RAI 23.01

F.1 PRA RAI 23.01

By letter dated April 23, 2013 (ADAMS Accession No. ML13116A015) the licensee responded to PRA RAI 23 and described plans to upgrade the fire HRA to NUREG-1921 during the National Fire Protection Association Standard 805 "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", 2001 Edition, (NFPA 805) implementation period, and to notify the NRC and provide a resolution plan if the upgrade causes the risk acceptance guidelines to be exceeded. Propose a regulatory mechanism that provides confidence that this upgrade will be completed. One regulatory mechanism used in NFPA-805 transition are the implementation items in LAR Attachment S, Table S-3. Since this is a PRA upgrade which could have a substantive impact on the PRA, the upgrade and a focused scope peer review should be completed prior to self approval.

OPPD Response to PRA RAI 23.01

OPPD will revise Table S-3 of the NFPA 805 LAR to include the following new implementation item:

"OPPD will upgrade its fire HRA following the methods described in NUREG-1921 EPRI/NRC-RES Fire Human Reliability Analysis Guidelines. The revised fire HRA will undergo a focused scope peer review against the HRA supporting requirements of ASME/ANS RA-Sa-2009, and resulting F&Os will be addressed. The NFPA 805 fire risk evaluations will be updated, and OPPD will notify the NRC should the risk acceptance guidelines be exceeded. OPPD will complete this implementation item prior to exercising the NFPA 805 self-approval process." **[AR 48249]**

Appendix H: PRA RAI 25

H.1 PRA RAI 25

Section 2.4.3.3 of NFPA 805 states that the PSA (PSA is also referred to as PRA) approach, methods, and data shall be acceptable to the AHJ. Section 2.4.4.1 of NFPA-805 states that the change in public health risk from any plant change shall be acceptable to the AHJ. RG 1.174, provides quantitative guidelines on CDF and LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plants licensing basis and describes a general framework for determine the acceptability of risk-informed changes.

The responses to the following RAIs provided sensitivity analyses to show the impact on fire risk of the indicated PRA modeling:

- **PRA RAI 01.c.01 regarding analysis of hot work-induced cable fires.**
- **PRA RAI 01.g.01 regarding cable routing assumptions for cables EB12191G and 7700A-B.**
- **PRA RAI 02 regarding main feedwater (MFW) pump oil fire severity factors.**
- **PRA RAI 07.01.c regarding combustible control violations.**
- **PRA RAI 16.b regarding modeling of the auxiliary feedwater (AFW) pump.**
- **PRA RAI 21 regarding qualitative screening of MCR heating ventilation and air conditioning (HVAC) scenarios.**
- **FM RAI 01.d.ii regarding fire modeling of vent opening size.**
- **FM RAI 01.e regarding fire modeling of open door cabinets with non-qualified cable.**
- **FM RAI 01.03 regarding inappropriate use of the McCaffrey, Quintiere, Harkleroad (MQH) method.**

The responses to the above RAIs indicate that these analyses individually and collectively do not have a potentially significant impact on the Δ risk for the transition based on the results of the individual sensitivity analysis. However, the self-approval guidelines are two orders of magnitude smaller than the transition acceptance guidelines and all future changes to the FPP must be adequately evaluated. For each of the above methods, identify which method is intended to be used in the PRA that will be used to support post-transition change in risk evaluations. Continued use of unacceptable methods will prohibit the staff from completing its review for self approval.

OPPD Response to PRA RAI 25

The following table summarizes which methods will be implemented in the post-transition fire PRA. **[AR 48249]**

RAI	Method to be used by Post-Transition Fire PRA
PRA 01.c.01	The post-transition FPRA will implement the FAQ 13-0005 methodology for cable fires caused by welding and cutting.
PRA 01.g.01	Cables EB12191G, 7700A, and 7700B were subject to cable routing assumptions in the base FPRA supporting LIC-11-0099. The actual field routing of these cables was determined in response to PRA 01.g.01, and knowledge of this routing slightly increased the calculated fire risk for specific scenarios. The post-transition FPRA will implement the actual field routing of cables EB12191G, 7700A, and 7700B.
PRA 02	The post-transition FPRA will implement the main feedwater pump oil fire severity factors specified by NUREG/CR-6850 Supplement 1 Chapter 9.
PRA 07.01.c	The post-transition FPRA will include contribution of combustible control process violations as described in response to PRA RAI 07.01.c.
PRA 16.b	Response to PRA RAI 16.b evaluates a second FW-10 demand that is not modeled and for which the pump could fail to start. The post-transition FPRA will model pump failure upon this second demand consistent with response to PRA RAI 16.b.
PRA 21	The post-transition FPRA will quantitatively model sequences involving loss of control room HVAC per response to PRA RAI 21.
FM 01.d.ii	The post-transition FPRA will model compartment-specific ventilation opening areas as described in response to Fire Modeling RAI 01.d.ii, rather than the one square meter generically assumed for all compartments by the base fire PRA supporting LIC-11-0099.
FM 01.e	The post-transition FPRA will quantitatively include risk contribution of open door electrical cabinets as described in response to Fire Modeling RAI 01.e.
FM 01.03	The post-transition FPRA will implement the method of Foot, Pagni, and Alvares (Equation 2-7 of NUREG-1805) for calculating upper layer temperature within mechanically ventilated compartments.

Appendix J: PRA RAI 27

J.1 PRA RAI 27

By letter dated May 21, 2013 (ADAMS Accession No. ML13144A814) the licensee responded to PRA RAI 22 and indicated that no Bayesian Update is necessary for those fires self-identified as potentially challenging. Three past plant events appear to be candidates for input into a Bayesian update. The response indicated that the control room fire is within the range of data considered by EPRI 1019259 (NUREG/CR-6850, Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements." However, the control room fire, dated 11/29/1997, is not in the database upon which the frequencies for NUREG/CR-6850 were established. Updating the generic bin 4 MCB frequencies from EPRI 1019259 with this event will make a substantial increase to the bin 4 fire frequency (approximately a factor of 2).

For the transient fire and electrical cabinet events, the update of each generic frequency will increase the frequency. Electrical cabinet fires include all events, at power and not at power. For bin 15.1, the addition of another event over 10 years will also approximately double the fire frequency for that bin. For bin 7, the increase is of a lesser amount, yet an increase will occur in frequency for bin 7 also. Thus, in each case, the plant has experienced more fires than would be expected from the generic industry database.

As a result, please perform a sensitivity study through updating the frequency bins for those events. Provide the impact on CDF/LERF/ Δ CDF/ Δ Delta LERF. More information may also be provided regarding the statement that the plant does not have a particular susceptibility to the fire type, as compared to the industry operating experience, and therefore the generic frequency is applicable.

OPPD Response to PRA RAI 27

The generic fire frequencies provided in NUREG/CR-6850 Supplement 1 consider industry fire event data through the year 2000. The period used for this Bayesian update is from 2001 through June 2013, or 12.5 years. Bayesian update to address the three subject FCS fire events is summarized in the following table.

Fire Event	Date	Bin	α_{prior}	β_{prior}	Posterior Mean
Control Board Fire ++(CR199701629)	11/29/1997	4	1	1212.9	1.63E-03 /yr.
Stressing Gallery Fire (CR200103787)	12/19/2001	7	5.03	1045.1	5.71E-03 /yr.
1B4A Load Center HEAF (CR 2011-5414)	06/07/2011	15.2*	1.5	1419.3	1.75E-03 /yr.

*Note that the 1B4A load center fire was originally classified as a Bin 15.1 event in response to PRA RAI 22. Upon further review, this event is more appropriately classified as a Bin 15.2 High Energy Arcing Fault (HEAF).

The following table summarizes the total plant CDF, total plant LERF, VFDR ΔCDF, and VFDR ΔLERF for the base fire PRA (Reference LIC-11-0099) and for the sensitivity study Bayesian update of the Bin 4, 7, and 15.2 fire frequencies.

	Base Fire PRA (LIC-11-0099)	**Sensitivity Study (Bayesian Update)**
Net VFDR ΔCDF for NFPA 805 Transition (/yr.)	5.72E-06	5.78E-06
Net VFDR ΔLERF for NFPA 805 Transition (/yr.)	6.67E-07	6.68E-07
Total CDF (internal, flood, fire) (/yr.)	6.01E-05	7.03E-05
Total LERF (internal, flood, fire) (/yr.)	4.82E-06	5.50E-06

In conclusion, the total CDF, total LERF, VFDR ΔCDF, and VFDR ΔLERF remain within RG 1.174 Revision 1 Region II when the Bin 4, 7, and 15.2 fire frequencies are Bayesian updated. Approximately 51% of the total CDF increase is attributed to Bin 15.2, approximately 44% is attributed to Bin 4, and approximately 5% is attributed to Bin 7. The total LERF increase was similarly distributed.

This is considered a conservative evaluation since the occurrence of only one event within each bin does not necessarily indicate a particular susceptibility of OPPD to those fire types, as compared to the industry operating experience. On the other hand, a repeated set of events (more than one) within a given bin might suggest a plant-specific vulnerability warranting Bayesian updating. If the generic frequencies are updated to include these OPPD events, contribution of the events will be distributed across the total reactor years for all US reactors, as opposed to only the OPPD reactor years (resulting in frequencies much lower than used in this assessment). Nonetheless, the risk metrics remain within the RG 1.174 Revision 1 Region II numerical acceptance guidelines when Bayesian update is performed.