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August 22, 2017
L-17-268

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

SUBJECT:
Beaver Valley Power Station, Unit Nos. 1 and 2
Docket No. 50-334, License No. DPR-66
Docket No. 50-412, License No. NPF-73
Response to Request for Additional Information and Supplemental Information Regarding License Amendment Request to Adopt National Fire Protection Association Standard 805 (CAC Nos. MF3301 and MF3302)

By letter dated December 23, 2013 (Accession No. ML14002A086), as supplemented by letters dated February 14, 2014; April 27, 2015; May 27, 2015; June 26, 2015; November 6, 2015; December 21, 2015; February 24, 2016; May 12, 2016; January 30, 2017; April 21, 2017; and June 23, 2017 (Accession Nos. ML14051A499, ML15118A484, ML15147A372, ML15177A110, ML15313A306, ML15356A136, ML16055A160, ML16133A340, ML17030A312, ML17111A883, ML17111A884, ML17111A885, ML17111A886, ML17111A887, and ML17177A097, respectively), FirstEnergy Nuclear Operating Company (FENOC) submitted a license amendment request (LAR) to change the Beaver Valley Power Station Unit No. 1 (BVPS-1) and Unit No. 2 (BVPS-2), fire protection program to one based on the National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition.

By correspondence dated August 8, 2017, the Nuclear Regulatory Commission (NRC) requested additional information to complete its review. The FENOC response is attached to this letter. The enclosure to this letter replaces LAR Attachment S, Table S-3, "Implementation Items," that was previously transmitted to the NRC in the April 21, 2017 letter.

The information provided by this submittal does not invalidate the significant hazards consideration analysis provided in the December 23, 2013 letter.

There are no regulatory commitments included in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager - Fleet Licensing, at (330) 315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 22, 2017.

Sincerely,

Richard D. Bologna

Attachment:
Response to Request for Additional Information

Enclosure:
LAR Attachment S, Table S-3, Implementation Items

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

The NRC staff's request for additional information (RAI) is provided in bold text followed by the FENOC response.

Probabilistic Risk Assessment (PRA) RAI 03.b.01

In its letter dated April 27, 2015 (ADAMS Accession No. ML15118A484) in response to PRA RAI 14, the licensee stated that the uncertainty analysis for fire-induced core damage frequency (CDF) and (large early release frequency (LERF) would be re-evaluated taking into account state of knowledge correlation (SOKC). In its letter dated June 23, 2017 (ADAMS Accession No. ML17177A097) the licensee responded to PRA RAI 03 and explained that though such an analysis was performed, the results were not incorporated into its integrated analysis risk results because the impact was determined to be "very small." Provide the following:

a. **The results of the analysis showing the increase in the risk estimates (i.e., core damage frequency (CDF), large early release frequency (LERF), delta (Δ) CDF and ΔLERF) from including the contribution of SOKC. Compare this increase to the Regulatory Guide (RG) 1.205, Revision 1, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (ADAMS Accession No. ML092730314) risk acceptance guidelines for self-approved changes to determine the magnitude of the impact.**

b. **If the impact of including the contribution from SOKC is not small compared to the RG 1.205 risk acceptance guidelines for self-approved changes, then add an implementation item to LAR Attachment S that provides for the incorporation of the contribution from SOKC into fire risk estimates used to support future self-approved changes.**

Response:

a. The increase in risk estimates per year when including the contribution of SOKC is no more than 0.3 percent of the reported CDF and LERF estimates for BVPS-1 and BVPS-2, and is shown in the following table:

Impact of SOKC on Net Delta Risk and Transitioning Plant Risk							
	Delta Risk Excluding Risk Offset Based on Point Estimate	Delta Risk Excluding Risk Offset - Allowing for SOKC Effects	Risk Offset	Net Delta Risk Allowing for SOKC Effects	Transition Plant CDF or LERF Based on Point Estimate	Transition Plant CDF or LERF Allowing for SOKC Effects	Total Transition Plant CDF or LERF Increase Due to SOKC Effects
BVPS-1 CDF	5.89E-05	5.91E-05	-9.72E-05	-3.81E-05	4.60E-05	4.61E-05	1.38E-07
BVPS-1 LERF	4.31E-07	4.32E-07	-6.44E-07	-2.12E-07	4.56E-07	4.57E-07	1.37E-09
BVPS-2 CDF	6.86E-05	6.88E-05	-7.20E-05	-3.19E-06	5.92E-05	5.94E-05	1.78E-07
BVPS-2 LERF	1.78E-06	1.79E-06	-2.67E-06	-8.85E-07	1.11E-06	1.11E-06	3.33E-09

From the far right column of this table, the effect of the SOKC can be seen to add approximately 1.38E-07 per year (/yr) to CDF and 1.37E-09 /yr to LERF for BVPS-1, and adds approximately 1.78E-07 /yr to CDF and 3.33E-09 /yr to LERF for BVPS-2.

b. Describing the risk increase of including the effect of SOKC as very small in the response to PRA RAI 03.b is consistent with RG 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," where a risk increase of less than 1.0E-06 /yr is characterized as very small. Even though these increases can be considered very small in the context of the overall plant risk, the effect of SOKC may not be insignificant when compared against the self-approval acceptance guidelines in RG 1.205 and therefore cannot be discounted. A new implementation item (BV1-3345) has been added to LAR Attachment S, Table S-3, stating that the impact of SOKC on delta CDF and delta LERF against the self-approval acceptance criteria for post-transition changes will be considered. An updated LAR Attachment S, Table S-3 is enclosed with this letter.

PRA RAI 19.01

In its letter dated June 23, 2017 (ADAMS Accession No. ML17177A097) in its response to PRA RAI 19, the licensee identified seven conservative modeling assumptions and stated that though these conservatisms increase the total risk, they are "*present in both the transitioning plant model and the compliant model and so they have no effect on the reported change in risk.*" The NRC staff does not agree with this statement since conservative assumptions in the transition plant model can impact the risk of a variance from a deterministic requirement (VFDR). The NRC staff found that there appears to be a number of VFDRs that

cannot be fully modelled because of conservative modeling assumptions that were made in the Fire PRA. For example, the modeling assumption to fail instrument air for all fires impacts calculation of the risk of VFDRs associated with loss of instrumentation air to atmospheric steam dump valves (ASDVs). If both the compliant and transition plant models assume that instrument air is failed then the change in risk calculated for fire scenarios in which the ASDVs are required to be opened using instrument air will be underestimated. Provide the following:

a. **Identification of the VFDRs for the plant whose risk contribution could be impacted by conservative modeling.**

b. **If conservative modeling is identified that can impact the calculation of risk associated with a VFDR and lead to underestimation 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. If the total risk increase associated with unresolved VFDRs is not offset by the total risk decrease associated with risk reduction modifications when the conservative modeling is removed, then provide updated total CDF, LERF, ΔCDF and ΔLERF risk estimates and demonstrate that the risk acceptance guidelines in RG 1.205 are still met.**

c. **As an alternative to the request in part (b), provide an updated integrated analysis that replaces the conservative modeling with realistic modeling that does not underestimate the total change-in-risk.**

Response:

a. VFDRs whose risk contribution could be impacted by conservative modeling are listed in Table 19.01-1.

b. The response to PRA RAI 19 stated there are seven conservative modeling assumptions present in both the transitioning plant model and the compliant model, and they have no effect on the reported change in risk. The intent of this statement was to affirm that such modeling assumptions are initially present in both the transitioning plant and compliant plant models. In other words, both models are initially identical, and only subsequently diverge with changes made to specifically account for the effects of VFDRs in the compliant plant model.

   For BVPS-1 and BVPS-2, the VFDR evaluation process involves reviewing the fire PRA model to ensure a complete understanding of how the effects of each VFDR are modeled. This includes careful consideration of any potentially relevant conservative modeling assumptions, and how they may impact the quantification of delta risk. A determination is made as to whether the model can appropriately characterize the delta risk contribution of the VFDR, or if changes must be made to the model in order to appropriately quantify the delta risk. Once these factors are considered, FENOC personnel then decide upon the most appropriate way to create the compliant plant model and estimate the associated delta risk. When a modeling assumption is identified that could potentially impact the delta risk quantification for a VFDR, the determination of the appropriate compliant case modeling includes consideration of how such an assumption must be altered, removed, or effectively bypassed, within the context of the VFDR evaluation, in order to allow proper quantification of a realistic delta risk value.

In response to this RAI, FENOC reviewed the modeling of VFDRs for potential impacts of modeling assumptions. This review confirmed that the transitioning and compliant plant models are treated appropriately to ensure that VFDR delta risk is properly characterized and is not underestimated. The specific results of this review are summarized in Table 19.01-1.

As shown in Table 19.01-1, the VFDRs determined to be potentially impacted by conservative modeling assumptions are those associated with the assumed failure of the instrument air system; specifically the VFDRs for failure of the atmospheric steam dump valves (ASDVs) at BVPS-1, as described in the RAI example (the ASDVs at BVPS-2 are electro-hydraulic and do not rely upon instrument air). This VFDR exists for fire compartments at BVPS-1 since the assumed failure of instrument air renders the ASDVs unable to open automatically or remotely for fires in the transitioning plant model, and a local operator action is required. When evaluating the compliant case for this VFDR, the initial process review identified the assumption of instrument air failure as potentially having an effect on the ability to estimate the delta risk. In recognition of this fact it was determined that the VFDR compliant case could not guarantee success of the basic events for the ASDVs since the instrument air support system dependency would continue to fail the ASDV function in the fault tree logic. Instead, it was determined that the compliant case would bypass consideration of instrument air relative to the ASDVs, essentially assuming instrument air is available to the valves, by guaranteeing success of the fault tree in which the ASDVs are modeled. Using this compliant case modeling, the delta risk for the VFDR will not be underestimated because the transitioning plant considers the automatic or remote function of the ASDVs to always be failed while the compliant plant considers the ASDVs function to always be successful. This results in reporting the maximum possible delta risk for this VFDR.

An aspect of this particular VFDR (also identified in the initial VFDR process review) is that without instrument air in the transitioning plant model, the main steam safety valves (MSSVs) will initially be relied upon to relieve steam pressure until an operator can locally open the ASDVs. The assumed loss of instrument air renders the normal decay heat removal paths (the ASDVs and residual heat release [RHR] valve, which functions identically to the ASDVs except that it can release steam from the three steam generators) inoperable automatically and remotely, in which case only the MSSVs remain available for initial decay heat removal. Since fire scenarios would therefore fail the normal decay heat removal capability, more consideration was given to modeling the MSSVs.

On further investigation it was determined that for the time required for a local operator to travel to and operate an ASDV or the RHR valve, the total number of times the MSSVs should be expected to lift and reseat (cycle) could be estimated for each steam generator. The total conditional probability for MSSVs on a given steam generator to fail to open during this event was then calculated and found to be very low, such that their risk contribution is negligible and they do not need to be explicitly modeled. In comparison, the probability of a MSSV to fail to reseat after lifting during this event was determined to be substantially higher, and this failure mode is included in the transitioning plant fire PRA models so that it could properly contribute to the delta risk of the steam release VFDR in question.

The consequence of a MSSV failing to reseat after opening would be an uncontrolled cooldown and an eventual safety injection (SI), and was added to the transitioning plant model. This failure mode is set to guaranteed success in the compliant plant model, in order to fully capture aspects of the VFDR and its associated delta risk. This additional failure mode is represented by basic event XXMSSVSI in the discussion of modeling conservatism item number one in Table 19.01-1.

c. The VFDR evaluation process at BVPS-1 and BVPS-2 entails a thorough review of the aspects of each VFDR relative to the fire PRA models in order to ensure that modeling assumptions are properly accounted for and the delta risk is not underestimated. A review of conservative modeling assumptions in the fire PRA models, relative to their potential impact on the evaluation of VFDR delta risk, has confirmed that modeling assumptions were considered and treated appropriately in the transitioning plant and compliant plant models such that the delta risk is not underestimated. Therefore, an updated integrated analysis is not required.

Table 19.01-1

Modeling Conservatism in Transitioning Plant Model	Systems / Function Supported	Related VFDR Type	Compliant Case Modeling Approach
1. Instrument air (IA) and containment instrument air are assumed to fail for fires, due to the low-temperature copper piping and soldered joints present throughout the system and in most areas of the plant, which would likely fail if exposed to fire temperatures	ASDVs - fail closed on loss of IA	Local operation of steam dumps due to loss of IA (BV1-2860)	The RHR valve is free of fire damage with instrument air available and manually operable from the control room. The remaining three ASDVs will remain closed. Successful operation of the RHR valve will provide an adequate steam relief path for decay heat removal and prevent cycling of the MSSVs, thus avoiding a potential spurious SI challenge. In the BV1RV5F2 fire PRA (FPRA) model this compliant case is modeled by making the following changes: XXMSSVSI=0.0 MSSV FAIL TO RECLOSE Additionally, top event CD is set to success for fires by guaranteeing the use of split fraction CDS. This ensures that cooldown (function of the ASDVs) will be successful, as long as auxiliary feedwater (AFW) is available. **DELTA RISK CONSIDERATIONS** In this case the potential for conservative transitioning plant modeling to provide an inadequate representation of the VFDR compliant case, due to the ASDVs' reliance on instrument air, was recognized and the compliant model was adjusted accordingly, by guaranteeing success of the top event CD which models the function to cooldown the secondary side using the ASDVs. The effect of forcing this guaranteed success is essentially the same as restoring the availability of instrument air to the ASDVs.

Modeling Conservatism in Transitioning Plant Model	Systems / Function Supported	Related VFDR Type	Compliant Case Modeling Approach
1. (continued)	Reactor coolant pump (RCP) thermal barrier cooling - Isolates on loss of IA	None. See modeling conservatism item 2 below	No related VFDRs, so no compliant case is required
	Main feedwater (MFW) - Isolates on loss of IA	None. See modeling conservatism item 3 below	No related VFDRs, so no compliant case is required
	Containment isolation valves - Fails to desired (closed) position	None. Air-operated containment isolation valves fail safe to their closed position.	No related VFDRs, so no compliant case is required
2. RCP thermal barrier cooling is not credited for fires, due to its dependence on instrument air	RCP seal cooling	None. There are no VFDRs for loss of RCP thermal barrier cooling. The delta risk associated with VFDRs related to the spurious re-initiation of thermal barrier cooling is addressed, and is not affected by this assumption (BV1-2526). Circuits which can spuriously re-initialize thermal barrier cooling could also cause it to fail initially, making this modeling assumption effectively redundant from the perspective of the VFDR.	No related VFDRs, so no compliant case is required

Modeling Conservatism in Transitioning Plant Model	Systems / Function Supported	Related VFDR Type	Compliant Case Modeling Approach
3. Main feedwater is assumed to fail for fires, except for the cases in which its continued operation would have a more severe consequence (the ability to trip the main feedwater pumps and condensate pumps is modeled)	None. AFW is the credited source for secondary cooling inventory. Potential spurious operation of MFW system is evaluated for impact on AFW.	None. MFW is not credited for Nuclear Safety Performance Criteria (NSPC) and so has no associated VFDRs, except for spurious failures in which MFW may fail to trip.	No related VFDRs, so no compliant case is required
4. Emergency alternating current (AC) power cross-tie between BVPS-1 and BVPS-2 is not credited, due to the time required to establish the off-normal alignment and other expected effects of a fire	Provides redundant source of power for most support and frontline systems in internal events PRA model.	None. Emergency AC power cross-tie is not credited for NSPC and so has no associated VFDRs.	No related VFDRs, so no compliant case is required
5. Quench spray pumps are not credited; however they are modeled for potential spurious operation	None. Design basis type accidents requiring success of the containment depressurization function of the quench spray system cannot be induced by a fire.	None. Quench spray pumps are not credited for NSPC and so have no associated VFDRs.	No related VFDRs, so no compliant case is required

Modeling Conservatism in Transitioning Plant Model	Systems / Function Supported	Related VFDR Type	Compliant Case Modeling Approach
6. Inside recirculation spray system (providing containment depressurization during the recirculation phase) is not credited. It is not modeled since spurious operation would have no consequence, aside from potentially cavitating and failing the pumps that are not credited.	None. Design basis type accidents requiring success of the containment depressurization function of the recirculation spray system cannot be induced by a fire.	None. The containment spray function of the recirculation spray system is not credited for NSPC and so has no associated VFDRs.	No related VFDRs, so no compliant case is required
7. RCPs are assumed to require deliberate action to trip, even if power is lost to the supporting 4160 volt bus – this accounts for potential inopportune timing of failures, such that the bus may not actually fail until later in the scenario after the RCP has continued to operate long enough to induce a	RCP trip function. The internal events PRA model takes credit for loss of offsite power initiating events to automatically trip the RCPs such that operator action will not be required. The fire PRA model acknowledges that the specific timing of fire-induced loss of offsite power busses is not known and therefore cannot be credited to	RCP breakers (BV1-2973) RCP breakers and pressurizer spray (BV1-2138) VFDRs are written for fire impacts which would prevent the RCPs from tripping via the benchboard controls. The modeling assumption does not underestimate delta risk because the transitioning plant model assumes the RCPs are initially running per normal plant operation and require manual action to trip, either from the control	RCPs trip function set to success by setting affected RCP supply breaker basic events to 0, and guaranteeing availability of DC support power in the RCP top event (OC). Operator action is always required, at least from the benchboard controls, because there is no automatic trip function. This assumption inherently guarantees that the delta risk of VFDRs regarding RCP trip is not underestimated, since the transitioning plant does not take credit for a fire-induced loss of 4160 volt busses to cause a default trip of the RCPs and instead assumes that operator action (either control room or local, depending on the particular fire effects) will always be required to trip the RCPs for fire scenarios. The compliant case assumes that support systems are available and the operator can trip the RCPs from the control room. Associated VFDRs are only written when the ability to trip the RCPs from the control room is

Modeling Conservatism in Transitioning Plant Model	Systems / Function Supported	Related VFDR Type	Compliant Case Modeling Approach
seal LOCA.	automatically trip the RCPs within the required time. Therefore the transitioning plant fire PRA model always requires operator action (either control room or local) to trip the RCPs.	room, or locally at the breakers if the controls or control power circuits are fire-affected. The compliant plant model assumes the remote controls and supporting DC control power for the RCPs are always available in support of the manual action to trip via the benchboard controls.	impacted by the fire.

PRA RAI 22.01

In its letter dated April 21, 2017 (ADAMS Accession No. ML17111A882), the licensee submitted a revised LAR Attachment S "Plant Modifications and Items to Be Completed."

a. **The descriptions of Implementation Items BV1-3108/BV2-1622 in LAR Attachment S, Table S-3, do not reflect the full set of actions provided in the licensee's letters dated April 27, 2015 (ADAMS Accession No. ML15118A484) and April 21, 2017 (ADAMS Accession No. ML17030A312). The NRC staff requests that the licensee revise Implementation Items BV1-3108/BV2-1622 to reflect the wording proposed in the letters dated April 27, 2015 and April 21, 2017:**

   > **Update FPRA Model, Risk Metrics, and Change-in-Risk Values following completion of risk-relevant modifications and implementation items; Update quantitative screening of fire scenarios in this model accounting for CDF and LERF in accordance with the criteria endorsed in RG 1.200, Rev. 2; and Implement new plant modifications or PRA refinements as necessary if the risk from the updated FPRA Model with the updated quantitative screening of fire scenarios exceeds the acceptance guidelines of RG 1.174.**

b. **The "LAR Section/Source" column in LAR Attachment S, Table S-3 has not been updated to reflect accurate sources including RAI responses for some of the implementation items. This was identified for Implementation Items BV1-3108, BV1-3109, BV2-1622, and BV2-1623. The staff found that these implementation items reference Attachments S and V when the sources of these implementation items is actually in licensee letters dated April 27, 2015 and April 21, 2017 in response to PRA RAIs 22 and 24. The NRC staff requests that the licensee update the "LAR Section/Source" column for all implementation items in LAR Attachment S, Table S-3 to reflect all applicable sources.**

Response:

a. The description for LAR Attachment S, Table S-3 Implementation Items BV1-3108 and BV2-1622 has been updated to state the following:

   > Update FPRA Model, Risk Metrics, and Change-in-Risk Values following completion of risk-relevant modifications and implementation items; Update quantitative screening of fire scenarios in this model accounting for CDF and LERF in accordance with the criteria endorsed in RG 1.200, Rev. 2; and Implement new plant modifications or PRA refinements as

necessary if the risk from the updated FPRA Model with the updated quantitative screening of fire scenarios exceeds the acceptance guidelines of RG 1.174.

An updated LAR Attachment S, Table S-3 is provided in the enclosure to this letter.

b. The "LAR Section/Source" column in LAR Attachment S, Table S-3 has been updated to reflect the applicable sources. An updated LAR Attachment S, Table S-3 is enclosed with this letter.

Enclosure
L-17-268

LAR Attachment S, Table S-3, Implementation Items
(4 pages follow)

Table S-3, Items provided below are those items (procedure changes, process updates, and training to affected plant personnel) that will be completed prior to implementation of the new NFPA 805 fire protection program. This will occur 12 months after NRC approval.

Table S-3 Implementation Items

Item	Unit	Description	LAR Section/Source, RAI
BV1-0714	1	Complete Penetration Seal Database	Att. A1
BV1-1624	1	Update Calculation 8700-DEC-3574, Circuit Failure Task 9	Att. V
BV1-1633	1	Update Uncertainty and Sensitivity Analysis Task 15* (See Note 1)	Att. V
BV1-2358	1	Develop the BVPS-1 Fire Safety Analysis (FSA)	4.7.1
BV1-2360	1 & 2	Update FPP procedures to reflect power block areas listed in Attachment I	4.1.3, FPE RAI 13
BV1-2371	1 & 2	Update fire brigade pre-fire plans and training materials. In addition, update pre-fire plans to specify appropriate engineering controls, use temporary dams when appropriate, and include monitoring of potentially contaminated fire suppression water	Att. A1, Att. E, 4.4.2, FPE RAI 13, RR RAI 02
BV1-2706	1	Current Transformer Secondary Open Circuit Analysis will be performed on CTs with no open circuit protection and a turns-ratio which exceeds 1200:5. Any CT identified as continuing to present a secondary fire risk will be modified in order to eliminate such risk	Att. B, SSD RAI 02
BV1-2823	1 & 2	Update administrative process/engineering controls to address future installation of wiring above a suspended ceiling in accordance with NFPA 805	Att. A1, Att. L, FPE RAI 13
BV1-2825	1 & 2	Beaver Valley NPO Implementation Plan	Att. D, FPE RAI 13
BV1-2826	1	Document results of walk down that was performed inside BVPS-1 reactor containment to evaluate fire detector spacing	Att. K and Att. T
BV1-2828	1	Revise procedure 1OST-33.13B, Deluge Valve Fire Protection System Instrument Test, to define actuation criterion	Att. A2
BV1-2833	1 & 2	Update the Fire Protection Program Change Evaluation associated with the hydraulic calculation for fire protection system water supply	Att. A1, Att. A2, FPE RAI 13
BV1-2902	1 & 2	Procedure update 1/2-ADM-1901 to add reference to NFPA 805	General Requirements, FPE RAI 13
BV1-2903	1 & 2	Procedure update 1/2-ADM-1902 to meet requirements of NFPA 805	Att. A1, FPE RAI 13
BV1-2904	1 & 2	Procedure update 1/2-ADM-1903 to add reference to NFPA 805	Att. A1, FPE RAI 13

Table S-3 Implementation Items

Item	Unit	Description	LAR Section/Source, RAI
BV1-2905	1 & 2	Procedure update 1/2-ADM-1904 to enhance combustible controls program	Att. A1, FPE RAI 13
BV1-2906	1 & 2	Procedure update 1/2-ADM-1905 to enhance combustible controls program	General Requirements, FPE RAI 13
BV1-2907	1 & 2	Procedure update 1/2-ADM-1906 to meet requirements of NFPA 805 and to designate TCEAs.	Att. A1, FM RAI 01(f), FPE RAI 13, PRA RAI 06(b)
BV1-2908	1 & 2	Procedure update 1/2-ADM-1900 to cite the NRC as the AHJ; to establish new compensatory actions appropriate with the level of risk created by equipment unavailability; to enhance controls on use and storage of flammable liquids per NFPA 30; to specify for new and replacement interior finishes to align with NFPA 101-2000 Section 10.2.3 and 10.2.7; to enhance controls on flammable and combustible materials; and to enhance controls of flammable gas	Att. A1, FPE RAI 08, FPE RAI 13
BV1-2909	1 & 2	Procedure update 1/2OM-56B.4A.A to enhance controls of combustible liquids	Att. A1, FPR RAI 13
BV1-2974	1 & 2	Verify the Reported Change-in-Risk Upon Completion of PRA-Credited Implementation Items* (See Note 1)	Att. V, 4.5.2.2, FPE RAI 13
BV1-2975	1	Fire Protection Safe Shutdown Response Procedures must be updated for NFPA 805 implementation	Att. V
BV1-2989	1 & 2	Beaver Valley will implement a Fire Protection Monitoring Program in accordance with the NRC approved version of FAQ 10-0059	Att. A1, Att. V, 4.6.2, FPE RAI 13
BV1-3018	1 & 2	Procedure updates associated with NFPA 30 requirements	Att. A1, FPE RAI 13
BV1-3019	1 & 2	Establish a minimum set of qualification criteria for proficiency associated with fire brigade qualifications	Att. A1, FPE RAI 13
BV1-3020	1 & 2	Fire Brigade Procedure changes related to NFPA 600 requirements	Att. A1, FPE RAI 13
BV1-3026	1 & 2	Procedure update related to NFPA 20 requirements.	Att. A1, FPE RAI 13
BV1-3027	1 & 2	BVPS-1 and BVPS-2 LAR Attachment G implementation activities	Att. G, FPE RAI 13
BV1-3041	1	Update fire barrier surveillance procedures to include newly defined fire compartments	Att. A2
BV1-3060	1 & 2	Review and Update HRA once Final Fire Procedures are written* (See Note 1)	Att. V, FPE RAI 13
BV1-3065	1 & 2	New NFPA 805 Control Procedures and Processes	4.7.2, FPE RAI 13
BV1-3066	1 & 2	Maintaining appropriate compensatory measures	5.5, FPE RAI 13

Table S-3 Implementation Items

Item	Unit	Description	LAR Section/Source, RAI
BV1-3108	1	Update FPRA Model, Risk Metrics, and Change-in-Risk Values following completion of risk-relevant modifications and implementation items; Update quantitative screening of fire scenarios in this model accounting for CDF and LERF in accordance with the criteria endorsed in RG 1.200, Rev. 2; and Implement new plant modifications or PRA refinements as necessary if the risk from the updated FPRA Model with the updated quantitative screening of fire scenarios exceeds the acceptance guidelines of RG 1.174.* (See Note 1)	PRA RAI 01(c), PRA RAI 22, PRA RAI 24
BV1-3109	1	Update of Westinghouse SDS modeling in the FPRA model, using NRC accepted failure models* (See Note 1)	PRA RAI 16(c), PRA RAI 22, PRA RAI 24
BV1-3117	1 & 2	Fire Modeling Personnel Qualification	FM RAI 05(a), FM RAI 05(b)
BV1-3122	1 & 2	Administrative Procedure Update for Seavan Radiological Activity	RR RAI 01
BV1-3123	1 & 2	Update procedures to minimize duration of open electrical cabinets	FM RAI 01(c)
BV1-3345	1 & 2	BVPS will consider the impact of SOKC on delta CDF and delta LERF against the self-approval acceptance criteria for post-transition changes	PRA RAI 03.b.01
BV2-0362	2	Update procedures to enhance guidance on containment and monitoring of potentially contaminated fire suppression water	Att. E
BV2-0487	2	Update the CO2 flow and time criteria for 2-CV-6	Att. A2
BV2-0619	2	Develop the BVPS-2 Fire Safety Analysis (FSA)	4.7.1
BV2-1020	2	Current Transformer Secondary Open Circuit Analysis will be performed on CTs with no open circuit protection and a turns-ratio which exceeds 1200:5. Any CT identified as continuing to present a secondary fire risk will be modified in order to eliminate such risk	Att. B, SSD RAI 02
BV2-1022	2	Perform Procedural Enhancements to Pre-fire Plans for 2-DG-1, 2-DG-2 Control Circuits Vent Fans	Att. A2
BV2-1157	1 & 2	Plant Boundary Definition and Partitioning Document will be enhanced to provide additional detail	Att. V, FPE RAI 13
BV2-1166	1 & 2	Minor documentation updates to BVPS-1 and BVPS-2 Control Room Fire Model Analyses	Att. V, FPE RAI 13
BV2-1169	1 & 2	Minor Documentation Updates to BVPS-1 and BVPS-2 Structural Steel Reviews	Att. V, FPE RAI 13

Table S-3 Implementation Items

Item	Unit	Description	LAR Section/Source, RAI
BV2-1182	1 & 2	Minor Documentation Update to Task 13, Seismic Fire Interaction Reviews	Att. V, FPE RAI 13
BV2-1294	2	Updates to Task 9 Calculation (10080-DEC-3575)	Att. V
BV2-1314	2	BVPS-2 procedure update 2OST-33.13I to include heat detection activation time	Att. A2
BV2-1345	1 & 2	Procedure revision to include fire door OS52-21 inspection verification	Att. A2, FPE RAI 13
BV2-1365	2	Fire Protection Safe Shutdown Response Procedures must be updated for NFPA 805 implementation	Att. S
BV2-1369	1 & 2	Hydraulic calculations need reconstituted for the water-spray systems in 2-PA-3, 2-RC-1, 2-SG-1N, and 2-SG-1S	Att. A2, FPE RAI 13
BV2-1372	2	BVPS-2 procedure update 2OST-33.21 to include heat detector activation time	Att. A2
BV2-1566	2	Add caution statement to Pre-Fire Plan for fire hose in 2-SB-4	Att. A2
BV2-1570	2	Hydrogen signage update needed for NFPA 55 requirements	Att. A1
BV2-1576	2	Update fire barrier surveillance procedures to include newly defined fire compartments	Att. A2 and Att. I
BV2-1580	2	Update Uncertainty and Sensitivity Analysis Task 15* (See Note 1)	Att. V
BV2-1622	2	Update FPRA Model, Risk Metrics, and Change-in-Risk Values following completion of risk-relevant modifications and implementation items; Update quantitative screening of fire scenarios in this model accounting for CDF and LERF in accordance with the criteria endorsed in RG 1.200, Rev. 2; and Implement new plant modifications or PRA refinements as necessary if the risk from the updated FPRA Model with the updated quantitative screening of fire scenarios exceeds the acceptance guidelines of RG 1.174.* (See Note 1)	PRA RAI 01(c), PRA RAI 22, PRA RAI 24
BV2-1623	2	Update of Westinghouse SDS modeling in the FPRA model, using NRC accepted failure models* (See Note 1)	PRA RAI 16(c), PRA RAI 22, PRA RAI 24
BV2-1750	1 & 2	Incorporate NUREG-2169 and NUREG-2178 to Unit 1 and Unit 2 fire compartments* (See Note 1)	PRA RAI 03(b)

Note 1: Items BV1-1633, BV1-2974, BV1-3060, BV1-3108, BV1-3109, BV2-1580, BV2-1622, BV2-1623, and BV2-1750 are to be completed by the end of the second Unit 2 refueling outage currently scheduled for April/May 2020 (2R21).