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June 26, 2014
L-14-209

10 CFR 50.90

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 Regarding License Amendment to Adopt Technical Specification Task Force Traveler 425 (TAC Nos. MF2942 and MF2943)

By correspondence dated October 18, 2013 (Accession No. ML13295A006), FirstEnergy Nuclear Operating Company (FENOC) submitted a license amendment request for the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS). The proposed amendment would modify the BVPS Technical Specifications by relocating specific surveillance frequencies to a licensee controlled program with the implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies." The proposed amendment is consistent with Nuclear Regulatory Commission (NRC)-approved Technical Specifications Task Force (TSTF) Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – Risk Informed Technical Specification Task Force (RITSTF) Initiative 5b," with certain proposed deviations.

By correspondence dated June 3, 2014 (Accession No. ML14133A069), the NRC requested additional information to complete the staff's review. FENOC's response to this request is attached.

There are no regulatory commitments established in this submittal. If there are any questions or 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 June 26, 2014.

Sincerely,

Eric A. Larson

Attachment: Response to June 3, 2014 Request for Additional Information

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

By correspondence dated October 18, 2013, FirstEnergy Nuclear Operating Company (FENOC) submitted a license amendment request for the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS). The proposed amendment would modify the BVPS Technical Specifications by relocating specific surveillance frequencies to a licensee controlled program with the implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies."

By correspondence dated June 3, 2014, Nuclear Regulatory Commission (NRC) staff requested additional information to complete its review. Each request for additional information (RAI) is presented in bold type, followed by the FENOC response.

RAI 1:
The finding pertaining to IF-D5-01, located in Table 2 of Enclosure B of the LAR, observes that the licensee uses outdated Internal Flooding Pipe and Tank Break frequencies for their Internal Flood (IF) assessment. The peer review team suggested that the licensee should update this Surveillance Requirement (SR) "to reflect more recent experience and should include plant specific experience." The licensee addressed this by stating that the latest IF model and focused peer review supersedes this finding and that the resolution is documented. Please describe the result of this SR in the updated PRA model and focused peer review and its associated resolution.

<u>Response</u>:
The finding pertaining to American Society of Mechanical Engineers (ASME) RA-Sb-2005 Supporting Requirement (SR) IF-D5 [Fact and Observation (F&O) IF-D5-01] was identified during the 2007 BVPS Probabilistic Risk Assessment (PRA) Self-Assessment to determine any gaps present between the BVPS PRA Revision 4 models and meeting the Capability Category (CC) II SRs in the 2005 version of the ASME PRA Standard Addendum B, as amplified by Regulatory Guide (RG) 1.200, Revision 1. The ASME PRA Standard SR IF-D5 deals with determining if the flood-initiating event frequency for each flood scenario group by using the applicable requirements in Table 4.5.1-2(c) was met. Two of the requirements in this table are to use the most recent applicable data to quantify the initiating event frequency and to account for plant availability (fraction of time the plant is at-power) in the initiating event analysis. F&O IF-D5-01 was assigned a Level B finding (instances where the issues have the potential to affect the risk results or insights) and was written since the BVPS PRA Revision 4 internal flooding assessments used pipe and tank break frequencies based on 1988 and 1990 reports that used data through September 1987. The finding recommended that the prior pipe break frequencies be updated to reflect more recent experience and should include plant specific experience.

In the BVPS PRA Revision 5a models, the internal flooding assessments were upgraded to meet the requirements of the ASME/ANS PRA standard RA-Sa-2009, along with the NRC clarifications and qualifications provided in RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." Therefore, based on the upgraded internal flooding PRA models, the basis for the F&O IF-D5-01 is no longer valid.

The source of the generic data for pressure boundary failures in the Revision 5a internal flooding PRA models are now taken from the November 2010 Electric Power Research Institute (EPRI) Technical Report 1021086, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments, Revision 2," and the companion Errata for Tables ES-2 and 6-3, December 2010 of this report. The generic data is based on piping system experience from 1970 through 2009. Plant-specific data was collected for the BVPS units from January 1, 1997 through June 30, 2010 for use in updating the pressure boundary failure/rupture frequencies estimated from the generic leakage/rupture rates given in the EPRI database, using the Bayesian technique.

These BVPS PRA Revision 5a models underwent a focused scope peer review for internal flooding in June 2011 using the process defined in NEI 05-04, Revision 2, "Process for Performing Interval Events PRA Peer Reviews Using the ASME ANS PRA Standard." Based on NEI 05-04, Revision 2, the former Supporting Requirement IF-D5 in the 2005 version of the ASME PRA Standard Addendum B, is now addressed in 2009 ASME Standard SR IFEV-A5. The 2011 BVPS Internal Flood PRA Focused Peer Review determined that SR IFEV-A5 was met with no F&Os.

RAI 2:
The finding pertaining to IF-D5-02, also located in Table 2 of Enclosure B of the LAR, observes that the licensee uses generic capacity factor data, which lowers the Initiating Event Frequency (IEF) and causes inconsistent IF IEFs for pipe break. The peer review team suggested that "the calculation for IF IEF be revised to be consistent with the focused Peer Review Facts and Observations (F&Os) as well as with the method used for other IEFs." The licensee addressed this by stating that the latest IF model and focused peer review supersedes this finding and that the resolution is documented. Please describe the result of this SR in the updated PRA model and focused peer review and its associated resolution.

Response:
The finding pertaining to ASME standard RA-Sb-2005 SR IF-D5 (F&O IF-D5-02) was identified during the 2007 BVPS PRA Self-Assessment to determine any gaps present between the BVPS PRA Revision 4 models and meeting the CC II SRs in the 2005 version of the ASME PRA Standard Addendum B, as amplified by RG 1.200, Revision 1. The ASME PRA Standard SR IF-D5 deals with determining if the flood-initiating event frequency for each flood scenario group by using the applicable requirements in Table 4.5.1-2(c) was met. Two of the requirements in this table are to use the most recent applicable data to quantify the initiating event frequency and to account for plant availability (fraction of time the plant is at-power) in the initiating event analysis. F&O

IF-D5-02 was assigned a Level C finding (suggestion on an alternative approach to achieve an objective but does not imply that the approach used is not sufficient to meet the supporting requirements at the stated grade). The Level C finding was written since the BVPS PRA Revision 4 internal flooding assessments were based on a generic 80 percent capacity factor, which could result in slightly lower flooding initiating event frequencies (IEFs) and was inconsistent with the method used to calculate other IEFs. It was recommended that the calculation for the flooding IEFs be revised to be consistent with the method used for other IEFs.

In the BVPS PRA Revision 5a models, the internal flooding assessments were upgraded to meet the requirements of the ASME/ANS PRA standard RA-Sa-2009, along with the NRC clarifications and qualifications provided in RG 1.200, Revision 2. Therefore, based on the upgraded internal flooding PRA models, the basis for the F&O IF-D5-02 is no longer valid.

The Revision 5a internal flooding PRA models now use BVPS plant specific critical hours corresponding to the time period from January 1, 1997 to June 30, 2010 in the calculation of the internal flooding IEFs, which is consistent with the method used for the calculation of other IEFs.

These BVPS PRA Revision 5a models underwent a focused scope peer review for internal flooding in June 2011 using the process defined in NEI 05-04, Revision 2. Based on NEI 05-04, Revision 2 the former Supporting Requirement IF-D5 in the 2005 version of the ASME PRA Standard Addendum B, is now addressed in 2009 ASME Standard SR IFEV-A5. The 2011 BVPS Internal Flood PRA Focused Peer Review determined that SR IFEV-A5 was met with no F&Os.

RAI 3:

Please describe, in more detail, how fire and seismic events will be assessed in terms of the NEI 04-10 guidance. The licensee discusses the use of a Seismic and Fire PRA in Section 3.0, "External Events Considerations," of Enclosure C of the LAR, and states that it has not been peer reviewed. However, the licensee indicated that they plan to use these models to quantify Surveillance Test Interval changes. NEI 04-10 states, in part, that "Plants implementing TSTF-425 shall evaluate their PRAs in accordance with [Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1[3]]." The statements made in the submittal indicated that the fire and seismic portions of the BVPS PRA have not been assessed against the PRA Standard referenced in RG 1.200, Revision 2.[4] For fire and seismic external events, please explain whether a qualitative or bounding (step 10), or detailed risk analyses (step 11) described in the NEI 04-10 guidance will be used. It should be noted that the PRA model for step 11 should meet the technical adequacy of RG 1.200, Revision 2

[3] ADAMS Accession No. ML070240001.
[4] ADAMS Accession No. ML090410014.

Response:
The NEI 04-10 Revision 1, provides guidance for the assessment of fire and seismic events using quantitative assessment methods if a plant has a seismic or fire PRA model, or qualitative screening. Currently, both BVPS Units have seismic and fire PRA models, but they have not been peer reviewed against the PRA Standard referenced in RG 1.200, Revision 2. Therefore, the fire and seismic events will be assessed using a qualitative screening analysis that would provide some indication of the impact of the surveillance test interval (STI) change on the results. If the qualitative information is deemed sufficient to support the acceptability of the STI change with respect to the fire and seismic risk, it will be used to provide the basis for the qualitative conclusions to the Independent Decisionmaking Panel (IDP). Based on the NEI 04-10 Revision 1 guidance, since only qualitative considerations are provided in this case, the impacts of the STI change on fire and seismic events will not be incorporated into the cumulative impacts.

However, in accordance with NEI 04-10 Revision 1 Step 10a, if this qualitative information is not deemed sufficient for each external contributor, then a bounding analysis (Step 10b) would be performed using the BVPS seismic and fire PRA models to provide some indication of the impact of the STI change on the results. Although the BVPS seismic and fire PRA models have not been peer reviewed, FENOC considers the BVPS external event PRA models of sufficient scope to provide valuable insights to qualitatively assess the seismic and internal fire risk associated with this risk informed application.

Additionally, the National Fire Protection Association (NFPA) 805 Fire PRA models for both units will be used to perform quantitative risk assessment sensitivity cases on the STI changes that can be adequately characterized by the fire PRA. These NFPA 805 Fire PRA models were peer reviewed in accordance with RG 1.200 Revision 2 and all F&Os have been resolved to meet CC II or above; however, not all of the credited plant modifications are installed, so they cannot be used as the sole basis for fire risk impact. Once all of the plant modifications are installed, these fire PRA models will be used to perform a quantitative risk assessment, when possible, to determine the internal fire hazard risk metric inputs for core damage frequency (CDF) and large early release frequency (LERF) associated with the STI change in place of the qualitative screening approach.

BVPS is also in the process of developing seismic PRA models for both units built in accordance with RG 1.200, Revision 2. These models are expected to undergo a peer review in December 2014 in accordance with RG 1.200, Revision 2, and will be revised as necessary to achieve CC II or above. Once all of the seismic peer review F&Os are resolved, and the plant modifications are installed, these seismic PRA models will be used to perform a quantitative risk assessment, when possible, to determine the seismic hazard risk metric inputs (CDF and LERF) associated with the STI change in place of the qualitative screening approach. However, prior to implementation of these seismic PRAs, they too will be used to perform sensitivity cases on the STI changes that can be adequately characterized by the seismic PRA.

RAI 4:
In the F&O related to SY-B1, located in Table 2 of Enclosure C of the LAR, the licensee resolved the peer review teams' unmet finding by using a different source of information than what was prescribed in the American Society of Mechanical Engineers (ASME) RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," PRA standard (ASME PRA Standard). The licensee further states that Westinghouse Commercial Atomic Power (WCAP)-16672-P was used because it "addressed the concerns that were raised regarding the consistency and correctness of the CCF [common-cause failure] events included in the NRC CCF database." Please provide the points of deviation between the WCAP and NUREG/CR-5497 and their potential impact on the risk result.

Response:
The level of significance for the F&O related to SY-B1, located in Table 2 of Enclosure C of the LAR, was classified as a suggestion F&O and not a finding, since the SY-B1 supporting requirement was determined to be met.

The suggestion pertaining to ASME standard RA-Sb-2005 SR SY-B1 (F&O SY-B1-01) was identified during the 2007 BVPS PRA Self-Assessment to determine any gaps present between the BVPS PRA Revision 4 models and meeting the CC II SRs in the 2005 version of the ASME PRA Standard Addendum B, as amplified by RG 1.200, Revision 1. The SY-B1 supporting requirement was reviewed and determined to be met using the guidance of RG 1.200, Revision 1, Appendix B, Table B-4 to identify the applicable NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guidance," technical elements, one of which was Technical Element DA-8. A review of Technical Element DA-8 in the 2002 BVPS PRA Peer Review, identified that it had an associated finding (F&O DA-06) that was not adequately resolved. The F&O DA-06 issue was related to using the PLG generic CCF database (circa 1989) exclusively without providing justification for not using newer data (circa 1995) provided in NUREG/CR-5497, "Common Cause Failure Parameter Estimations." NUREG/CR-5497 was issued in October 1998, along with a CD-ROM disc for utilities to use in developing better common cause MGL parameter estimates.

When the 2007 Self-Assessment review of SR SY-B1 was performed it determined that the supporting requirement was met with a CC II/III rating, since the BVPS Revision 4 PRA models did include intra-system common cause failures when supported by generic or plant-specific data, using a process that follows the guidance set forth in NUREG/CR-5485, "Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessments." However, F&O SY-B1-01 was written as a Level C suggestion to improve documentation, since the BVPS Revision 4 PRA models did provide justification for continued use of the PLG Database, but the justification did not include a review of up-to-date data sources (such as NUREG/CR-5497).

NUREG/CR-5497 was also to be used in a Westinghouse Owner's Group project to develop a database consisting of a common set of realistic, generic common cause failure events to be used as the basis for generic and plant-specific MGL parameter estimations among the owner's group member utilities. During this project development it was noted that there were some major discrepancies in the classification of common cause failures between the published NUREG/CR-5497 and the CD-ROM. As such, the NRC was to update the coding of their common cause failure events and database, and Westinghouse was to provide input and feedback so that there would be consistency between plant-specific PRA models when using the data.

The Pressurized Water Reactor (PWR) Owners Group (PWROG) has conducted several programs in support of improving CCF data analysis for its member utilities. The initial program was conducted to address concerns that were raised regarding the consistency and correctness of the CCF events included in the NRC CCF database. To address certain shortcomings a systematic process was developed to review and assess the events in the CCF database, which covered the 1980 to 2000 timeframe. The review of the database events was based on criteria and guidelines that were established to define CCF events that were potentially applicable to the PWROG utilities. As part of the collection and verification process, the NRC requested the support of the PWROG in reviewing the CCF events to determine their technical accuracy prior to entry into the CCF database. Consequently, in addition to the initial program, the PWROG has supported the NRC in reviewing the CCF events that were collected since 2000. CCF events for the 2001 to 2003 timeframe were reviewed by Westinghouse on behalf of the PWROG. Based on the reviews conducted by Westinghouse for the CCF events collected during the 2001 to 2003 timeframe and interactions with Idaho National Laboratory, coding changes to certain CCF events were made that resulted in improvements to the accuracy and pedigree of the CCF database. The CCF database now includes events that cover the 1980 to 2003 timeframe.

The purpose of Westinghouse Commercial Atomic Power Report (WCAP)-16672-P, Revision 1, which was issued in June 2008, was to provide CCF parameter estimates for the events contained in the NRC's CCF database covering data from 1980 through 2003. These estimates covered the majority of risk-significant equipment that is important to safety at US nuclear power plants with Nuclear Steam Supply Systems designed by Westinghouse Electric Company or Combustion Engineering for eleven different systems.

During the BVPS Revision 5 PRA model updates, a decision was made to use the most up-to-date published information available for CCF parameter estimates, which were provided in WCAP-16672-P, Revision 1 and were based on the 2003 updated events in the NRC CCF database. These estimates were recommended for use by the applicable PWROG utilities instead of the parameter estimates provided in NUREG/CR-5497, which had not been revised to reflect the occurrence of CCF events since 1995. The use of the generic WCAP CCF data does not diminish the SR SY-B1 CC II/III rating, since the requirement to model intra-system common cause failures when supported by generic or plant-specific data using a process that follows the guidance set forth in NUREG/CR-5485, is still met.

There are no significant deviations between the WCAP and NUREG/CR-5485. NUREG/CR-5485 provides guidelines that PRA analysts can use for modeling CCF, along with a set of generic alpha factors (that is, not component/system specific) with no operating data, which may be used when a more detailed evaluation is not feasible. To this, the WCAP CCF mapping process did follow the guidance provided in NUREG/CR-5485 and also developed a set of generic alpha factors for components with no operating data that are reflective of PWR CCF operating experience. These values were found to be comparable to the set of generic alpha factors, which are provided in NUREG/CR-5485. As such, it is expected that there would be no significant impacts on the risk result by using the WCAP generic CCF data compared to the NUREG/CR-5485 generic CCF data.

Additionally, a CCF sensitivity case was performed at each of the BVPS units using the 95 percentile CCF values to quantify the CDF. The results of these sensitivities revealed that there were no significant impacts on the CDF; the Unit 1 CDF increased by 3.1 percent, while the Unit 2 CDF increased by 4.4 percent.

RAI 5:
The submittal highlights the licensee's PRA model changes and peer reviews against the ASME PRA standard. The ASME PRA Standard and RG 1.200, Revision 2 clarify the definition of a model update versus an upgrade. Please provide clarification on the model changes as to whether they were updates or upgrades.

Response:
The list summarizing the BVPS-1 PRA model revision history provided in Section 2.3, Applicability of Peer Review Findings and Observations (F&Os), of Enclosure B of the license amendment request (LAR) submittal is recreated below and includes information as to whether the PRA model change was an update or an upgrade.

Date	Revision	BVPS-1 PRA Model Change	PRA Update or Upgrade
10/1992	0	Individual Plant Examination (IPE) NRC submittal	N/A
06/1995	1	Individual Plant Examination – External Events (IPEEE) NRC submittal	N/A
06/1998	2	Integrated Level 1 and Level 2 models	Update
09/2003	3	WOG NEI 00-02 Peer Review with Category A/B F&Os addressed	Update
06/2006	4	HRA [Human Reliability Analysis] Calculator, replacement steam generators, atmospheric containment conversion, and extended power uprate model	Upgrade to HRA with 2007 Focused Scope Peer Review, Update for all other changes.

Date	Revision	BVPS-1 PRA Model Change	PRA Update or Upgrade
12/2010	5	RG 1.200, R1 (excluding Floods) CCII Compliant Model	Update
01/2013	5a	Interim model update to include Internal Flooding, RG 1.200, R1 (including Floods) CCII Compliant Model	Upgrade to Internal Flooding with 2011 Focused Scope Peer Review, Update for all other changes.

The list summarizing the BVPS-2 PRA model revision history provided in Section 2.3, Applicability of Peer Review Findings and Observations (F&Os), of Enclosure C of the LAR submittal is recreated below and includes information as to whether the PRA model change was an update or an upgrade.

Date	Revision	BVPS-2 PRA Model Change	PRA Update or Upgrade
03/1992	0	Individual Plant Examination (IPE) NRC submittal	N/A
09/1997	1	Individual Plant Examination – External Events (IPEEE) NRC submittal	N/A
10/1997	2	Integrated Level 1 and Level 2 models	Update
01/2002	3A	WOG NEI 00-02 Peer Reviewed	Update
05/2003	3B	WOG NEI 00-02 Peer Review with Category A/B F&Os addressed	Update
04/2007	4	HRA [Human Reliability Analysis] Calculator, atmospheric containment conversion, and extended power uprate model	Upgrade to HRA with 2007 Focused Scope Peer Review, Update for all other changes.
12/2010	5	RG 1.200, R1 (excluding Floods) CCII Compliant Model	Update
08/2012	5a	Interim model update to include Internal Flooding, RG 1.200, R1 (including Floods) CCII Compliant Model	Upgrade to Internal Flooding with 2011 Focused Scope Peer Review, Update for all other changes.

RAI 6:
The licensee discusses all of the PRA model changes and subsequent peer reviews in the submittal but does not provide any information regarding the resolution of the F&Os that were considered to be documented. Please provide those applicable F&Os and their resolution.

Response:
The BVPS F&Os and their final resolutions related to both the PRA modeling and documentation issues from the PRA peer reviews, focused scope peer reviews, and self-assessments, were previously provided to the NRC on February 14, 2014 as Supplemental Information Regarding Application for License Amendment to Adopt NFPA 805, "Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) (Accession Number ML14051A499).